[redacted]

ARC-100 – Alternate Shutdown System – Why it is Not Needed
White Paper

Document Number: CSRS-004_Rev 1.0
Date: January 28 2025

[redacted]

ARC NOTICES

[redacted]

[redacted]

[redacted]

[redacted]

REVISION CONTROL SHEET

Revision	Date	Description
0A	October 29 2024	First Issue.
0B	January 17 2025	Revised to modify Safety Rod configuration, and addition of control rod drive mechanism diversity.
1.0	January 28 2025	Final Revisions.

[redacted]

TABLE OF CONTENTS

[redacted]

LIST OF TABLES

LIST OF FIGURES

[redacted]

Acronym and Terms / Definitions

Acronym	Term / Definition
ACLP	Above Core Load Pad
ADAMS	Agencywide Documents Access and Management System
ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrence
ARC	Advanced Reactor Concepts
ARDC	Advanced Reactor Design Criteria
B_4C	Boron Carbide
BDBE	Beyond Design Basis Event
BOL	Beginning of Life
CCCG	Common Cause Component Group
CCF	Common-Cause Failure
CDEHO	Component, Design, Environment, Human, and Other
CFR or C.F.R.	Code of Federal Regulations
cm	centimeter
CRA	Control Rod Assembly
CRBR	Clinch River Breeder Reactor
CRDM	Control Rod Drive Mechanism
DBE	Design Basis Event
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EHO	Environment, Human, and Other
EOL	End of Life
°F	Degrees Fahrenheit
FFTF	Fast Flux Test Facility
FOAK	First of a Kind
FPGA	Field-Programmable Gate Array
FR	Federal Register
GDC	General Design Criteria
GEM	Gas Expansion Module
HALEU	High Assay Low Enriched Uranium
HT9	A ferritic martensitic stainless-steel alloy that's known for its High-Temperature strength and resistance to irradiation damage
INL	Idaho National Laboratory
IVTM	In-Vessel Transfer Machine
Li^6	Lithium atom with 3 protons and 3 neutrons
LIM	Lithium Injection Module
LLC	Limited Liability Company
LWR	Light Water Reactor
MGL	Multiple Greek Letter
mm	millimeter
MWt	Megawatt-thermal
NRC	U.S. Nuclear Regulatory Commission
NUBOW-3D	a computer program for the static three-dimensional analysis of bowed reactor cores
NUREG	NUclear REGulatory – NRC technical report designation

Acronym	Term / Definition
OD	Outer Diameter
OUO	Official Use Only
pcm	percent mille - a unit of reactivity, one one-thousandth of one percent (or, 1×10^{-5})

[redacted]

EXECUTIVE SUMMARY

The purpose of this white paper is to provide ARC Clean Technology's (ARC's) position that an alternate reactor shutdown system, that is not depending on rods, is not necessary to meet regulatory requirements. The white paper shows that the two independent rod based - Primary (Control) and Secondary (Safey) rod systems have separate diverse drive systems, redundant and diverse actuation systems, each backed by a second system that uses the drive motors to if necessary force the rods into the core, when combined with the highly reliable reactor protection system, have an extremely low probability of failing to shut down the reactor.

To supplement and reinforce the position that the two rod systems are sufficient the white paper also addresses what systems have been proposed by others, pointing out the similarity of their approach to ours, as well as providing a summary of the review performed by the IAEA on possible alternative shutdown systems.

Core deformations sufficient to impede the insertion of the rods has been also determined to be extremely unlikely, based on development of the deformations that can be expected in the core, the gap designed between the control elements interior ducts and the control assemblies exterior duct; and the means to anticipate unacceptable deformations by comparing pre-calculated forces generated by the deformation, with periodically measuring the force to extract the control and fuel assemblies, and if necessary rotating the assemblies to reverse the deformation.

The white paper seeks the Nuclear Regulatory Commission opinion on whether the information presented herein is sufficient to establish a high confidence that an additional system, not based on rods, is required. Nevertheless, should the arguments presented herein be seen as insufficient, this white paper also presents a concept that could be utilized to provide an alternate shutdown system.

[redacted]

1. INTRODUCTION

The purpose of this white paper is to provide ARC Clean Technology's (ARC's) position that an alternate reactor shutdown system, that is one not depending on rods, is not necessary to meet regulatory requirements.

This position is predicated on the very high reliability of:

- two separate rod-based system,
- their drive and actuation system,
- the reactor protection system,
- the backup system that is designed to drive the rods into the core, and
- the extremely unlikely possibility that core deformation could develop (and not be identified in time) to an extent that rod insertion might be precluded.

Each of the above is addressed in detail in the body of the white paper.

In addition, a possible concept that could be utilized as an alternate, non-rod based system to shut down the reactor is also presented

1.1 Background

Under the provisions of 10 CFR §50.34, an application for a construction permit to build a nuclear production facility must include the principal design criteria (PDC) for a proposed facility. The PDC establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. 10 CFR Part 50, Appendix A provides General Design Criteria (GDC), which establish minimum requirements for the principal design criteria for light-water-cooled nuclear power plants (LWRs) of similar design to currently licensed LWRs. To provide similar guidance to non-LWR designers, applicants, and licensees the NRC developed Regulatory Guide (RG) 1.232 "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors" (Reference [8]).

Regulatory Guide (RG) 1.232, Appendix B provides design criteria specifically for sodium fast reactors (SFR DC). Of interest to this white paper is SFR DC 26 for reactivity control systems, which has been reproduced Table 1-1.

Table 1-1: SFR DC 26 — *Reactivity control systems*

SFR DC 26 — *Reactivity control systems*. A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal

power changes to assure that the design limits for the fission product barriers are not exceeded.

(3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.

(4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.

In addition to providing the SFR specific DC, RG 1.232, Appendix B provides the rationale for the changes that were made to each of the 10 CFR Part 50, Appendix A GDC. Appendix A of this white paper reproduces in full the NRC's rationale for the changes made to GDCs 26 and 27 to develop SFR DC 26. The changes that are the most pertinent to this white paper and how they affect the ARC-100 are briefly summarized below; a detailed description of how the ARC-100 complies with SFR DC 26 is provided in the following sections.

RG 1.232 has combined Part 50 GDC 26 and 27 into a single SFR DC 26, and deleted DC 27. The RG 1.232 SFR DC 26 and NRC rationale for adoption are provided below.

- RG 1.232 SFR DC 26 does not require the reactor be brought to cold shutdown, but rather it requires the reactor be brought to "safe shutdown". SECY-94-084 (Reference [9]) describes "the characteristics of a safe shutdown condition as reactor subcriticality, decay heat removal, and radioactive materials containment." For an LWR SECY-94-084 defines the safe and cold shutdown coolant temperatures as 420 °F and 200 °F, respectively. For the ARC-100, safe and cold shutdown will be 420-435 °F, and 180-200 °F, respectively. The range is provided since the pools at safe shutdown will initially equilibrate at a slightly higher temperature before they cool down.
- RG 1.232 states: "A minimum of two reactivity control systems or means shall provide…" RG 1.232 added "'means' in recognition that advanced reactor designs may not rely on control rods to rapidly shut down the reactor (e.g., alternative system designs or inherent feedback mechanisms may be relied upon to perform this function)."
    - Thus, ARC-100's negative reactivity feedback can be considered as one means of reactivity control. (see Section 2.1)
    - The ARC-100's negative reactivity feedback is one means of reactivity control and the primary control system is the second. This was the approach taken in the PRISM PSID (Reference [12]), which NRC concluded in the PSER (NUREG-1368 (Reference [13])) (see Appendix C.1):

        > *The GDC 26 requirement for an independent and diverse means of reactivity control is provided in the PRISM design by the inherent reactivity feedback of the design which, according to the designers, brings the reactor to zero power upon loss of flow or loss of a normal heat removal path, even if there is a failure to scram. This is acceptable to the staff as a means of meeting GDC 26 and the minimum level of safety criteria …, provided that*

certain conditions can be met (…). Adequacy of the proposed design to meet the purpose of this GDC through passive feedbacks should be demonstrated by prototype testing before the design certification stage. (Reference [13], p 3-36)

- Nevertheless, it is germane to note that inherent reactivity feedback does not bring the reactor to a subcritical state, but only a controlled stable and safety state with the reactor still critical
- Additionally, either the primary or secondary rods can independently shut down the reactor. Therefore, if sufficient diversity exists in the rod, the latching mechanism, the actuation mechanism, and the drive mechanisms, the safety and control rods may be considered as two independent and diverse systems. (see Section 2.2)

[redacted]

2. ARC-100 REACTIVITY CONTROL MEANS

This section describes the current ARC-100 means for reactivity control, which include 1) inherent reactivity feedback, 2) a primary (control) rod system, and 3) a secondary (safety) rod system. Then it describes the ARC-100 Reactor Protection System (RPS) and the Idaho National Laboratory (INL) prepared probability safety analysis which determined the probability of the RPS not shutting down the reactor. Core deformations and the rod deformation analysis that was performed by Argonne National Laboratory (ANL) are described next. Finally, a determination that an alternate shutdown system is not required is presented.

2.1 Inherent Reactivity Feedback

The ARC-100's inherent reactivity feedback controls the fission process within the core. As temperatures rise, the net negative feedback reduces power. The natural feedbacks are self-regulating, inherent in the fuel design, and result in a safe and stable power level at which heat production and heat removal are in balance.

The primary reactivity feedback mechanisms in increasing order of response time are:

- Doppler feedback: Effect of changes in neutron fission and absorption cross sections due to Doppler broadening
  - Negative for ARC-100 at elevated temps
- Coolant density and void worth: Effect of changes in Na coolant atom cross sections
  - At elevated temperatures, this could be positive due to reduced Na absorption, or negative due to enhanced neutron leakage
- Axial fuel expansion: Effect of thermal expansion of oxide/metal fuels in the cladding tube
  - Negative at elevated temperatures due to reduced number density of fissionable isotopes
- Radial core expansion: Due to thermal expansion, irradiation-induced swelling, and irradiation-enhanced creep
  - Negative at elevated temperatures due to enhanced leakage

Following an anticipated operational occurrence (AOO), a design basis event (DBE), or a Beyond Design Basis event (BDBE),the ARC-100 must be placed into one of the two (2) following states:

- Safe state: the reactor is subcritical, long-term heat removal is assured, radioactive releases are controlled, and acceptable and reactor coolant inventory is acceptable.

- Controlled state: an inherent and passive state in which long-term heat removal is assured, radioactive releases are controlled and acceptable and reactor coolant inventory is acceptable, and the reactor in not subcritical.

In the controlled state the ARC-100's inherent reactivity feedback will self-regulate itself to a power level equal to the amount of heat being transferred out, without control or safety rod movement. The ARC-100 can remain in a controlled state for an indefinitely long period, without any fuel clad breaching. Thus, the ARC-100 controlled state meets the first part of SFR DC 26(1) condition (i.e., the design limits for the fission product barriers are not exceeded),

[redacted]

[redacted] it does not meet the second condition (i.e., safe shutdown is achieved and maintained).

2.2 Reactivity Control System

The ARC 100 reactivity control system consists of two independent redundant and diverse absorber rods systems, either one of which can shut down the reactor: the primary (or control) rods and the secondary (or safety) rods.

2.2.1 Primary (Control) Rods

The Primary Control Rod System provides shutdown, reactivity and power control through six independently regulated absorber assemblies (control rods). Each control rod unit consists of a drive mechanism, a driveline, and a control assembly (absorber bundle and absorber channel). The control assembly is shown in Figure 2-1. The absorber bundle is a closely packed array of seven tubes containing compacted natural boron carbide (B_4C) pellets. The tubes, or “pins”, are each helically wrapped with wire and bundled into a triangular pitch, hexagonal pattern. The wire wrap maintains the pin spacing so that coolant may circulate freely through the pin bundle. The bundle of pins is contained in a duct that channels flow through the bundle and protects the pins from damage as they slide within the outer fixed duct. The pin and duct assembly material is HT9 stainless steel. The locations of the six primary control assemblies in the core pattern are shown in Figure 2-2.

The primary control rods provide start-up control, power control, burnup compensation, and absorber run in (“runback”) in response to demands from the plant control system. These control rod movements are accomplished with the rod shim motors. This system is diverse from that employed for the Secondary Safety Rods and is a roller nut-which is the same as that proposed for the Clinch River Breeder Reactor (CRBR) and used at the Fast Flux Test Facility (FFTF).

The system also provides rapid shutdown in response to demands from the Reactor Protection System (RPS). RPS-directed control rod scram is accomplished by releasing two (2) electromagnets (solenoids) per rod which releases the absorber into core. The RPS is a four-division safety related (Safety Classification 1) system, using a combination of analog sensors and digital solid-state circuity (field-programmable gate array (FPGA) logic) to perform the logic decision. Each of the four divisions, using its own transducers, measures a scram parameter, compares it to a setpoint digitally and appropriately issues a trip demand. If two out of the four divisions issue a trip demand, the reactor is scrammed. In this case, the scramming mechanism de-energizes the two solenoids of each primary control rod and thereby dropping them into the core by gravity. The scram requires that both solenoids be de-energized per primary control rod which allows surveillance testing without actually scramming. Following a scram the shim motors will activate to follow the gravity drop of the rods and provide sufficient force to insert the rod if the control assembly duct has suffered deformation.

It is important to note that the absorber pins are contained within a separate hexagonal duct (as shown at the top left of Figure 2-1), and that hexagonal duct is separated from the external assembly duct by a gap of about 9 mm. This means that a significant amount of deformation would be necessary before the ducts could bind and fail to insert (see Section 2.4).

[redacted]

[redacted]

Figure 2-1: Primary Control Rod

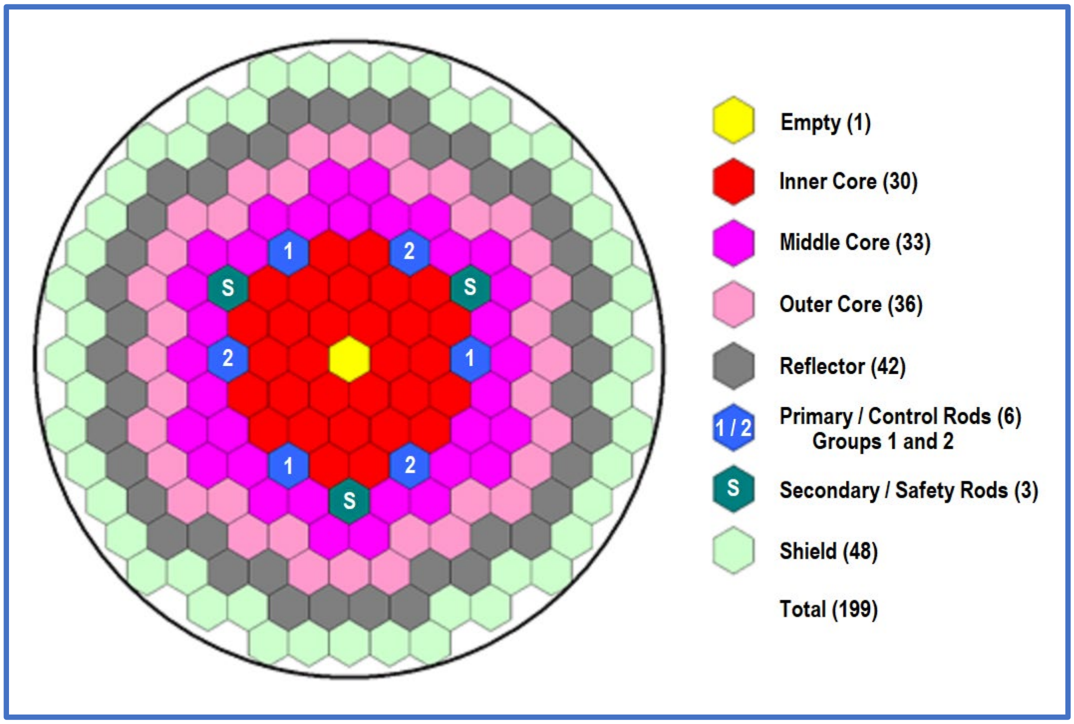


Figure 2-2: ARC-100 Core Configuration

As shown in Figure 2-2, the six primary control rods are arranged into two groups with three rods each. To introduce diversity into the RPS it is specified that the load drivers and solenoids will each be obtained from at least two (2) different manufacturers, Primary Control Rod Group 1 are from manufacturer alpha, while Group 2 are from manufacturer beta. Specifying different manufacturers will reduce (but not eliminate) the probability of a common cause failure (see Appendix B). For example, having different manufacturers will reduce design and manufacturing flaws, it will not reduce flaws caused by environmental conditions, human errors, or other causes.

2.2.2 Secondary (Safety) Rods

The Secondary Safety Rod System provides a backup shutdown method through three independently controlled absorber assemblies. The Secondary Safety Rod System does not regulate power, during power operation they are fully withdrawn and parked above the core.

Initially the absorber element of the safety rods had been designed to be essentially identical to the primary control rod elements. For diversity this has been changed to a different configuration, as described in Appendix D. The Secondary Safety Rod system design is now diverse from the Primary Control Rod design in three ways.

a. The absorber element is configured differently from the control rod absorber. While the control rod have a total of 7 identical pins of natural boron carbide, contained within an inner hexagonal duct, the safety rods use 8 pins of 40% enriched boron carbide, with seven pins having equal diameter and a central pin a greater diameter, contained within a cylindrical sheath which has an average gap of 11 mm and a minimum gap of 6.73 mm to the inner surface of the external hexagonal duct. The safety rod absorber configuration is shown in Figure 2.3

This diversity reduces the likelihood that both control rods and safety rods would share the same difficulty in insertion in the core.

b. The safety rod drive mechanisms are also different from those of the control rods. The control rod drive mechanism is a roller-nut type used in the Fast Flux Test Reactor (FFTF) and planned to be to be used in the Clinch River Breeder Reactor (CRBR). The safety rod drive mechanism is a rack and pinion type used at the EBR II.
c. Another difference is the manner (gripping mechanism) in which the rods are attached and disengaged to their drivelines.

The Secondary Safety Rod absorber material is compacted 40% enriched B_4C pellets clad in HT9 material, and the pin and duct assembly material is HT9 steel. The locations of the three secondary control assemblies in the core pattern are shown in Figure 2-2.

The Secondary Safety Rod System provides only for shutdown should the primary control rods fail to achieve shutdown. The Secondary Safety Rod System is not utilized for power control and its rods are fully withdrawn and parked above the core. The reactivity worth of the three secondary safety rods is sufficient to shut down the reactor to a safe state.

2.2.3 Control and Safety Rod Drive Mechanisms

The control rods drive mechanism is planned to be a roller nut type similar to that planned for CRBR and used in FFTF (the difference between the two being spring assisted fall vs. gravity fall off the rod), and the safety rods drive mechanism is planned to be a rack and pinion type as employed in EBR II. Figure 2.4 illustrates the two types. For further diversity our plan is to have the two different types manufactured by different companies.

For the rack and pinion system, no decision has yet been made on whether an air accumulator to force the rods into the core in case of electrical failure, as was the case at the EBR II and as shown in Figure 2-4, or to rely on the RTNSS electrical supply to drive the motors.

[redacted]

[redacted]

Figure 2-3: Secondary (Safey) Rod

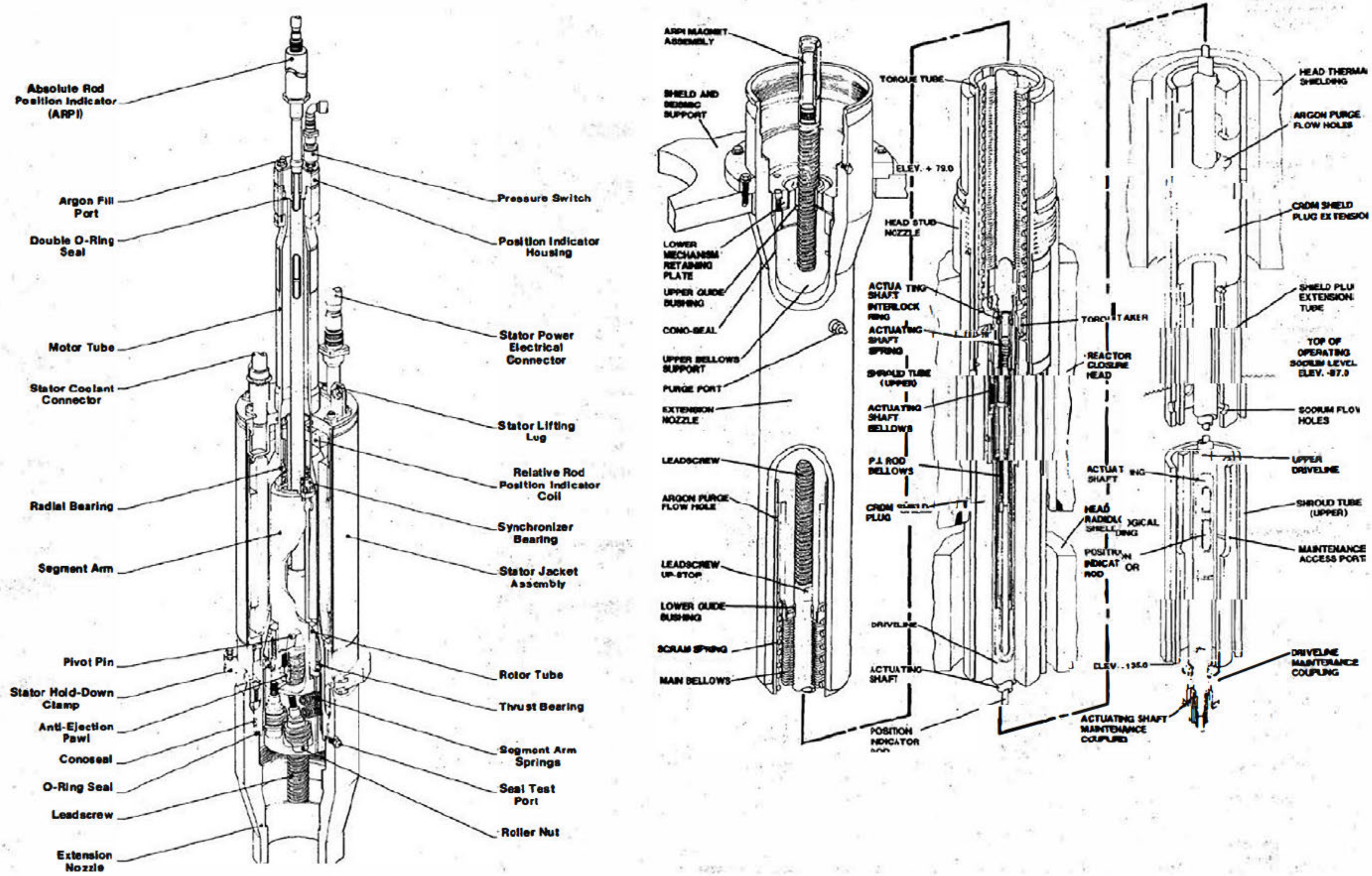


Figure 2-4 A Primary Control Rod Upper and Lower Roller-Nut Mechanism from CRBR [22]

[redacted]

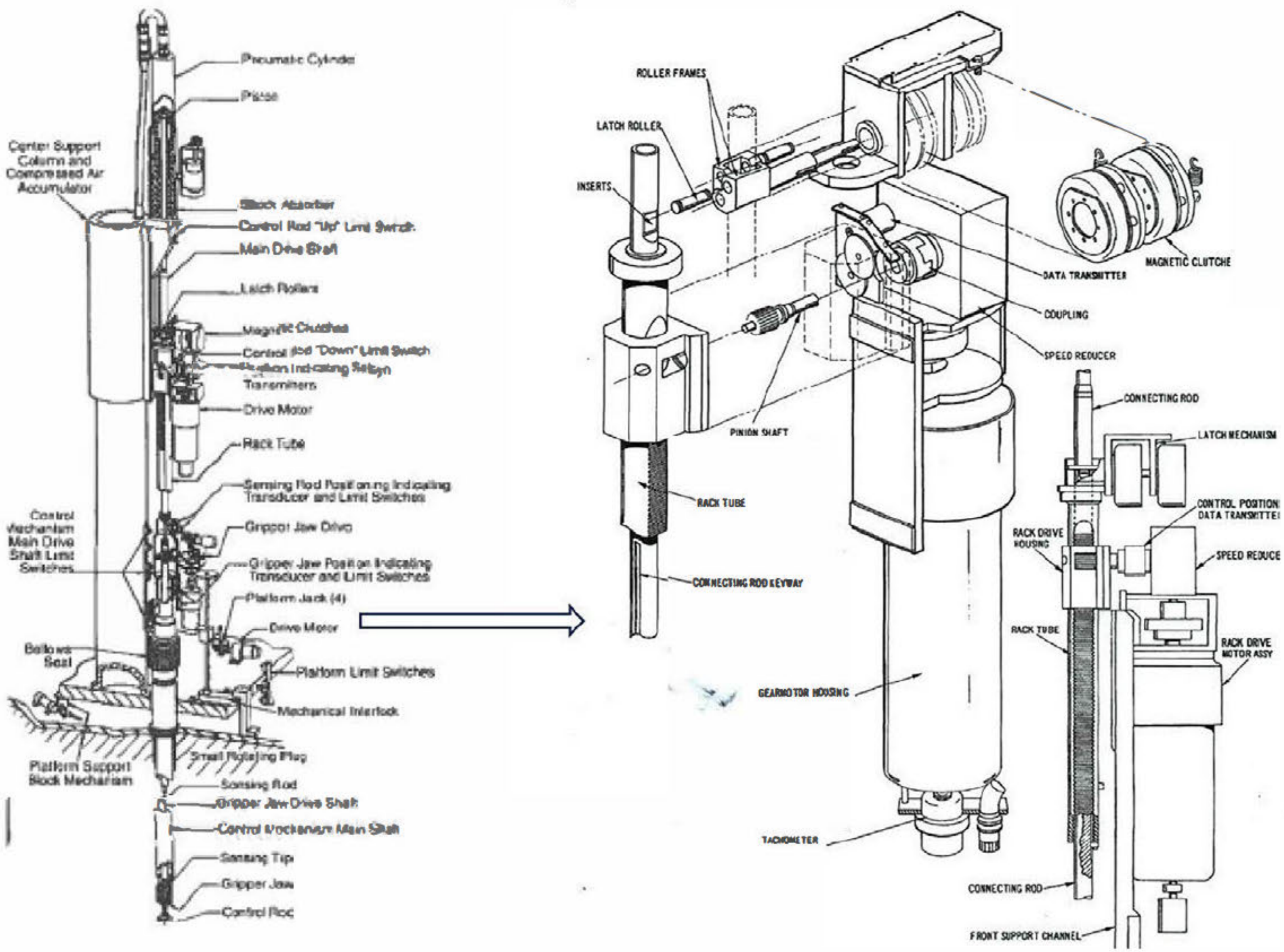


Figure 2.4 B Rack and Pinion Drive for Safety Rod (From EBR II - with air accumulator backup) [21]

2.3 Reactor Protection System Probability Safety Analysis

Reference [15] calculated the probability / frequency of ARC-100 RPS failure to shutdown the reactor. Two (2) cases were analyzed: 1) it was assumed that the primary control rod with the maximum worth had been withdrawn / ejected prior to the signal to activate the RPS and 2) it was assumed that upon receiving the signal to activate the RPS all primary control rods were in their normal BOL operating positions. For the first and second cases RPS failure probabilities / frequencies were calculated to be 1.371×10^{-5} (not including the probability of rod withdrawal / ejection, which for EBR-II was estimated to be 3.3×10^{-2} yr^{-1} for a combined probability of 4.52×10^{-7} and 6.175×10^{-8} $demand^{-1}$, respectively.

Some of the more important considerations made in the Reference [15] analysis are:

1) The ARC-100 Control Rod System and RPS designs are as they were in the summer of 2023. Although the ARC-100 preliminary design continues to evolve, significant modifications to the Control Rod System or RPS designs are not expected.
2) When determining the RPS failure probability both independent and common-cause failures (CCFs) were considered.
3) Individual component failure probabilities were obtained from INL/EXT-21-65055 [16] and its associated Excel file.
4) CCFs were determined using the alpha factor methodology as implemented in the SAPHIRE computer code [17] and using the alpha factors from [18].
5) CCF alpha factors were adjusted to reflect the diversity of some load drivers and solenoids that originate from a different manufacturer.
6) The rod with the maximum reactivity worth was always conservatively assumed to be the rod withdrawn / ejected and/or the first rod to fail to insert.
7) The analysis was conservatively performed using the BOL rod worths and shutdown worth requirement. At MOL and EOL the rods have more worth and the shutdown worth requirement is less.

For case 1, with one primary control rod withdrawn / ejected, SAPHIRE determined that the minimal cut set list contained 645 cut sets. Twelve (12) of the cut sets contributing 99.6% of the total RPS failure probability, i.e., all of the remaining (645-12=) 633 cut sets contribute about 0.4% to the total RPS failure probability. Each of the top 12 cut set contributors to the RPS failure probability is a single event CCF cut set.

For case 2, with all primary control rods in their normal operating positions, SAPHIRE determined that the minimal cut set list contained 140 cut sets. Unlike for case 1, for case 2 there were no single event cut sets, and each cut set included at least one CCF event. For comparison to case 1, the top 12 case 2 cut sets contributed 45% of the total RPS failure probability.

More information regarding the RPS probability safety analysis, focusing on CCFs, is provided in Appendix B, as well as Reference [15].

[redacted]

2.4 Core Deformations

The design of the control and safety rods, within a hexagon in turn contained in the control assembly ,with a gap of 9 mm, minimizes the possibility that the rods could bind and fail to insert by gravity or by the follow up of the drive motors.

A detailed analysis of the deformation that can occur in the core has been conducted and reported in the Final Report on Predictive Analysis of PRD as a function of Deformation and Other Anomalies [19]

The deformation that can be experienced by the assemblies in the core, including the outer ducts of control assemblies, have been calculated by using the NUBOW-3D Code. NUBOW-3D can evaluate individual assembly deformation and full-core analysis by estimating the full core response to input temperatures at multiple time points throughout the reactor life. Time steps are added to account for any changes in temperature and irradiation fast flux, which is included for life-time creep and swelling calculations. These calculations are critical to evaluating the core restraint system design and performance. The assemblies are considered as thin-walled hexagonal ducts modeled as beam segments in 3D space with associated geometric properties for bending. Inter-duct contact is modeled as ideal contact at the load pad face centers without friction or assembly twisting.

The reference core and restraint designs were used for preliminary analysis of the core restraint system parameters. Figure 2-5 shows the core assembly mapping and numbering used in NUBOW-3D for the bowing analysis. Beyond the outer (green) row of shield assemblies there are restraint rings at both the Top Load Pad in the assembly upper fitting (TLP) and Above Core Load Pad (ACLP) elevations with an initial gap of 0.50 mm. The results are shown in the figures below.

[redacted]

Y

[redacted]

Figure 2-5: Core assembly mapping with NUBOW-3D numbering for analysis

Figure 2-6 shows the contact behavior at the ACLP at BOL and EOL cases, with the blue circles representing scaled contact force values. As shown, the inner core region does not have contact and therefore lock-up is not guaranteed over the core lifetime, which suggested some changes to the initial core restraint design are desired. Those changes have not yet been made, but will be made in the detailed design phase of the project.

[redacted]

[redacted]

Figure 2-6: Contact at ACLP elevation for the reference core design for the BOL (left) and EOL (right) cases

The residual bowing deformation at the end-of-life condition for assembly removal and refueling of the reference case is shown in Figure 2-7. From the figure it is clear that there can be significant permanent deformations due to inelastic deformations over time with residual contact occurring between multiple rows of assemblies.

[redacted]

[redacted]

Figure 2-7: Elevation view of residual bowing deformation from center of core to outer core showing permanent assembly bowing and contact during refueling. Line 20 is a control rod assembly

The maximum deformation (generally inward) of the fuel assemblies has been computed to be between 0.29 and 0.31 cm at BOL and EOL respectively, however the control rods experience lesser deformation. as shown in Table 2-1 Therefore, it is unlikely that the core would deform sufficiently to cause a binding of the control rods.

Table2-1: Maximum Deflections of Control and Safety Rods Assembly Ducts at BOL, MOL and EOL

	BOL		MOL		EOL	
	Disp X (mm)	Disp Y (mm)	Disp X (mm)	Disp Y (mm)	Disp X (mm)	Disp Y (mm)
[redacted]						

[redacted]

To provide further assurance that core deformations would not prevent rod insertions, the pull-out force for assemblies, particularly those near the control and safety rods, will be periodically measured, and compared to the forces predicted by NUBOW-3D.

2.5 Rod Deformation Analysis

An additional potential cause of RPS failure is fuel assembly and/or control rod deformation. Such deformation could prevent the control rods from inserting into the core. This potential RPS failure mode is addressed in section 2.3. In this section it is noted that the force calculated by NUBOW-3D and shown in Figure 2.8, are utilized to size the motors to drive in the rods, in case gravity or spring assisted fall is insufficient to insert the rods.

Moreover, as stated in Section 2.4, to minimize the potential for deformation preventing the rods from inserting the ARC-100 operator will periodically use the In-Vessel Transfer Machine (IVTM) to reach and grab the control assemblies to determine the force required to move them. If the required force is greater than a pre-set limit, indicating a growing deformation, actions can be taken by rotating the assembly to even out any deformation.

[redacted]

[redacted]

Figure 2.8 Estimated upper bound peak withdrawal loads (without assembly weight) [lb] for fuel and control assemblies for base core restraint case

2.6 Determination that an Alternate Shutdown System is Not Required

The probability a that either the control rods or the safety rods cannot be inserted because of failure of the reactor protection system to actuate the rods or core deformation preventing the insertion is extremely small.

Combined with the inherent behavior of ARC-100 which brings the reactor to a stable, safe, but still critical state (controlled state) which enables actions to be take over time, rather than immediately and urgently, such as an example the operator manually de-energizing both solenoids in the rod drive mechanism, argues that another system , alternative to rods, Is not required.

The addition of such a system, even if possible, introduces an unnecessary complexity in the safety systems of the ARC-100, and in addition its inadvertent actuation possibly causing operational problem.

[redacted]

ARC Clean technology seeks the Nuclear Regulatory Commission's opinion in this matter.

Nevertheless, for completeness, we address system that could be provided.

3. Potential Alternate Shutdown Schemes

The choice of the proposed alterative shutdown scheme has been made after reviewing the world experience with such schemes, including what has been proposed for reactors similar to the ARC 100 and by other SMRs.

3.1 Shutdown Systems Alternatives

A review of the world's sodium cooled reactor industry's efforts to develop shutdown systems that are alternate to absorber rods has identified a limited number of alternatives that could apply to the ARC-100 FOAK. Reference [1] reports on the progress of several systems that have been studied to passively actuate shutdown in fast reactors, with some currently in active development for use in existing and future reactors. The systems are listed in Table 0-1 and the reasons they are not considered for the ARC-100 reactor is also given.

Table 4-1: (Alternate) Passive Shutdown Schemes

Shutdown System	Reason for Adoption or Rejection
Lithium expansion modules	**Rejected**: Such system can introduce both positive and negative reactivity (added complexity and possible safety issue), whereas the injection system can only insert negative reactivity
Lithium injection modules	**Adopted**: With modification to insert solid balls of Enriched boron or Li^6 carbide, instead of liquid. Avoids vacuum in core, and inserted poison can be retrieved by refueling equipment after rods are inserted. The solid poison insertion as been shown to be sufficiently effective, (see also Levitated Absorber Particles- below) however the delivery system is only a concept requiring detailed design and testing
Curie point latches	**Rejected**: This system only differs in actuation (which is passive) and relies on rods. Thus, actuation is diverse (but ARC already has diverse actuation – however both diverse systems are active), but sufficient deformation in the core could prevent the rods form inserting.
Thermostatic switches	**Rejected**: Same issue as above, the advantage of having multiple switches distributed all over the core does not outweigh the lack of diversity in the physical insertion of the poison, which is still via rods.
Lyophobic capillary porous systems	**Rejected**: Complex, but reliable system which basically equates to a spring that inserts the poison. Rejected because deformation of the duct through which the poison is inserted can defeat the force of the spring. ARC uses a motor driven force to insert the rods if for whatever reason they fail to insert by gravity. The difference is that the motor force is an active system whereas the lyophobic capillary porous system is passive.

[redacted]

Table 4-1: (Alternate) Passive Shutdown Schemes

Shutdown System	Reason for Adoption or Rejection
Flow levitated absorbers	**Rejected**: This provides a passive actuation of the rods but suffers from the same issue as others above and below, namely duct deformation can prevent the insertion of rods.
Cartesian divers	**Rejected**: Requires a path for the adsorber to float down or up into the core, based on a gas changing the pressure in the liquid containing the neutrally buoyant absorber. Requires communication of the gas to the primary system pressure, which introduces complexity, and path can still be precluded by deformation.
Levitated absorber particles	**Possible Adoption**: This is similar in principle to flow levitate rods, but uses flow levitated balls instead of rods, and hence it is less susceptible to deformations. This system is receiving considerable attention and modification have been proposed that combine some of the features of the Lithium Injection Module (LIM), with the concept of particle levitation. In the upper region, above the active core, spherical neutron absorbing boron carbide particles are placed. In case of overpower and loss of coolant transients, a seal will melt. The absorber balls are then no longer supported and fall into the active core region, inserting a large amount of negative reactivity. It is sufficiently interesting that ARC will continue to study it. At present, however ARC has developed a concept that drop the boron carbide particles in the form of balls into an empty assembly in the center of the core.
Enhanced thermal elongation of control rod drivelines	**Rejected**: Relies on additional rods. So, diversity from rod system is lacking.
Autonomous reactivity controls	**Rejected**: Such system could be housed in a special assembly in the center position of the ARC-100 core (which is presently empty). This sealed system consists of two reservoirs, located at the top and bottom of the special assembly, connected by two concentric tubes that link the reservoirs. The inner tube is open at both ends, while the outer tube is only open at the bottom. During operation the upper reservoir is completely filled with an expansion liquid, while the lower reservoir contains the same expansion liquid, and floating on top of it an immiscible absorber liquid. For transients that raise the coolant temperature above a limit to be determined) the expansion liquid in the upper reservoir expand in the inner tube, compresses the lower expansion liquid which raises the adsorber liquid into the core, causing a reduction in power and temperature. As the core cools down, the temperature of the expansion liquid starts falling, lowering the level of the adsorber in the core. So, this system will cause the reactor to reach a stable but critical condition, which is already assured by the ARC 100 by inherent means, and therefore it is not suitable for an alternate shutdown system.

[redacted]

Table 4-1: (Alternate) Passive Shutdown Schemes

Shutdown System	Reason for Adoption or Rejection
Travelling wave reactor thermostats	**Rejected**: Has the same limitation as the one above. It ensures passive achievement of a stable safe, but still critical condition, and hence it is not suitable as an alternate shutdown system.
Thermo-siphon based passive shutdown systems	**Rejected**: This system would employ three separate liquids of vastly different densities on top, within and below the bottom of the core, one of which being the absorber, descends irreversibly into the core as a result of core thermal imbalance. ARC rejection is based on the problematic recovery that would be caused by ruptures in the liquid systems.
Static absorber feedback equipment	**Rejected**: This is just another form of rod insertion and as such it is not diverse.

[redacted]

Table 4-1: (Alternate) Passive Shutdown Schemes

Shutdown System	Reason for Adoption or Rejection
Gas Expansion Modules	**Rejected**: Gas expansion modules (GEMs) were developed during DOE's Advanced Liquid Metal Reactor (ALMR) program to accommodate the alternative oxide core. A GEM located at the active core periphery produces a void on the periphery of the core when primary coolant flow is stopped for any reason. This voiding provides sufficient negative reactivity insertion during a loss of primary flow without scram transient to meet the NRC established safety criteria to bring the core subcritical. During the ALMR program the GEH PRISM team investigated the addition of GEMs of the reference metal core design under the Reduction of Core Sodium Void Worth work in the early 1990s. The conclusion of that work then was the addition of GEMs to the metal core is not warranted because the probability of voiding is already sufficiently low with the reference core configuration; the additional cost and complexity were not justified. A GEM is essentially a passive device for inherent shutdown which inserts negative reactivity during an unprotected loss of coolant flow in a primary system. The device is basically a hollow removable subassembly sealed at the top and open at the bottom. The gas trapped inside the subassembly expands when core inlet pressure decreases owing to flow reduction, which expels sodium from the subassembly. Neutron leakage increases and negative reactivity is inserted. By using a GEM, additional negative reactivity feedback is induced in the core by an increase of the neutron leakage, caused by the lowering of the coolant level due to the decrease of the coolant pressure at the core inlet under the loss of flow conditions. However, a GEM produces negative reactivity only when hydraulic pressure is lost. FFTF tested GEMs and the results were mixed. One of the reasons for ARC's rejection of this concept is that the integrity of the envelope of the GEM must be assured in order to avoid ingress of gas into the core and the consequent positive reactivity insertion. Presence of the GEMs can also limit the flexibility to control the reactor with flow.

PRISM has a shutdown system very similar to the ARC-100 Initially it proposed an Ultimate Shutdown System very similar to what we propose (B_4C balls), NRC had reviewed it and opined it may not be necessary. Later in time a revitalized Ultimate Shutdown System was proposed in combination with GEMS. Appendix B.1 provides more information on the PRISM approach.

For Natrium, TerraPower is proposing a shutdown system based on rods, with diversity provide by one rod subsystem being gravity inserted, and the other being inserted by motors However it is dependent on rods only. In their review, NRC staff generally found this approach to be [redacted] in meeting the underlying intent of DC 26 but could not make a final determination without further analysis. (Reference [11]) More details on Natrium's approach are provided in Appendix B.2.

The Hermes non-power reactor also uses rods for shutdown and has geometric differences in the two types of rods (control and shutdown), which we do not have. Geometric difference such as proposed by Hermes are difficult to implement in the compact geometry of a fast sodium cooled reactor. The Hermes approach is discussed in more detail in Appendix B.3.

4. POSSIBLE ACR-100 ALTERNATE SHUTDOWN SCHEME

At present the only concept being considered is a system to deliver balls of enriched boron carbide into the empty center assembly in the core. Neutronic analysis has shown the reactivity worth of the balls is sufficient to shut down the reactor and maintain in in shutdown conditions However the delivery of the balls into the assembly is a concept whereby balls stored in a liquid sodium tank with argon cover gas located above the CRDMs are allowed to fall by gravity through a perforated guide channel of diameter equal to the outlet of the empty assembly' The perforate channel would be supported in the Upper Internal Structure. Actuation is manual, because the reactor itself is in a stable controlled state at whatever power is wheeled out at the time., so no rapid actuation is required. Furthermore, it is assumed that this system would be manually actuated after the power to the primary and secondary pump is terminated , and the primary pump have completed their 2 minutes coast down, so the reactor will inherently reduce power to decay heat .Prevention of the argon gas in the tank from reaching and penetrating into the core is achieved by the perforation of the channel. In the reactor cover gas region, the tank argon would mix with the cover gas argon. The system, after detailed design, will require testing to determine the diameter of the balls that will minimize the possibility that balls would agglomerate during their descent into the core.

4.1 Reactivity of the Boron Carbide Balls and Sufficiency for Reactor Shutdown

The material in this section summarizes the analyses and results of an Export Controlled document [3] where the details of the work can be found.

The ARC-100 core [4] has been designed to have sufficient reactivity shutdown margin using two independent reactivity control systems: primary and secondary control systems. To provide an additional redundant and diverse means to bring the reactor deeply subcritical in the case of failure of all control rods, an alternative shutdown system is considered: an injection of B_4C balls into the empty duct located in the center of the core.

The empty duct is assumed to be the same as the normal fuel assembly duct, but the active fuel and above are open to injection of B_4C balls. The negative reactivity inserted by the alternative shutdown system is dependent on various parameters, such as the ball diameter, the thickness of the ball cladding, packing density of the balls in the duct, enrichment of boron, etc. In the work of Reference [3], the reactivity worth of the alternative shutdown system was evaluated in terms of ball diameter, boron enrichment, and number of balls.

The Argonne Reactor Computation (ARC) code suite was used to evaluate the reactivity worth of the alternate shutdown system. Configuration with B_4C balls was modeled as a homogenized mixture within the active core axial section of the central duct. Region-dependent 33-group

[redacted]

[redacted] cross sections were generated using the MC^2-3 code. The core reactivity was calculated using the DIF3D code at the beginning of life (BOL) because the ARC-100 core has the highest excess reactivity at BOL. The negative reactivity worth is the reactivity difference with and without the B_4C balls in the central duct.

Ball packing density and material volume fractions are major factors in designing the alternative shutdown system. The ball packing density is defined by the volume fraction of balls in the duct where balls are inserted.

It is surprisingly difficult to specify the ball packing density because balls are somewhat randomly distributed when balls are poured into the empty duct. To approximate the random packing, Reference 3 adopted a regular hexagonal lattice arrangement of the balls in the duct. Figure 4-1 shows a single layer of the regular hexagonal lattice arrangement. The number of balls in a single hexagonal lattice layer is dependent on the ball diameter and the duct inner wall flat-to-flat distance. The alternative shutdown system can stack hexagonal lattice ball layers up to the top of the active core. Thus, the total number of balls of the alternative shutdown system can be calculated using the number of balls in a single hexagonal layer and the number of hexagonal lattice layers. Then, the ball packing density can be calculated using the total number of balls.

It is noted that the regular hexagonal lattice arrangement is one of the configurations to approximate the random packing density in a hexagonal duct. From the literature, the packing density for random packing in an infinite region is around 60% [5]. The packing density will necessarily be smaller in the bounded hexagonal duct depending on the ball diameter. It is noted that in the Appendix, a subsequent calculation performed independently from that of Reference 3 for a stack of balls with an OD of 3.95 cm (39.5 mm), determined that a packing fraction of the B_4C balls could be about 60%.

[redacted]

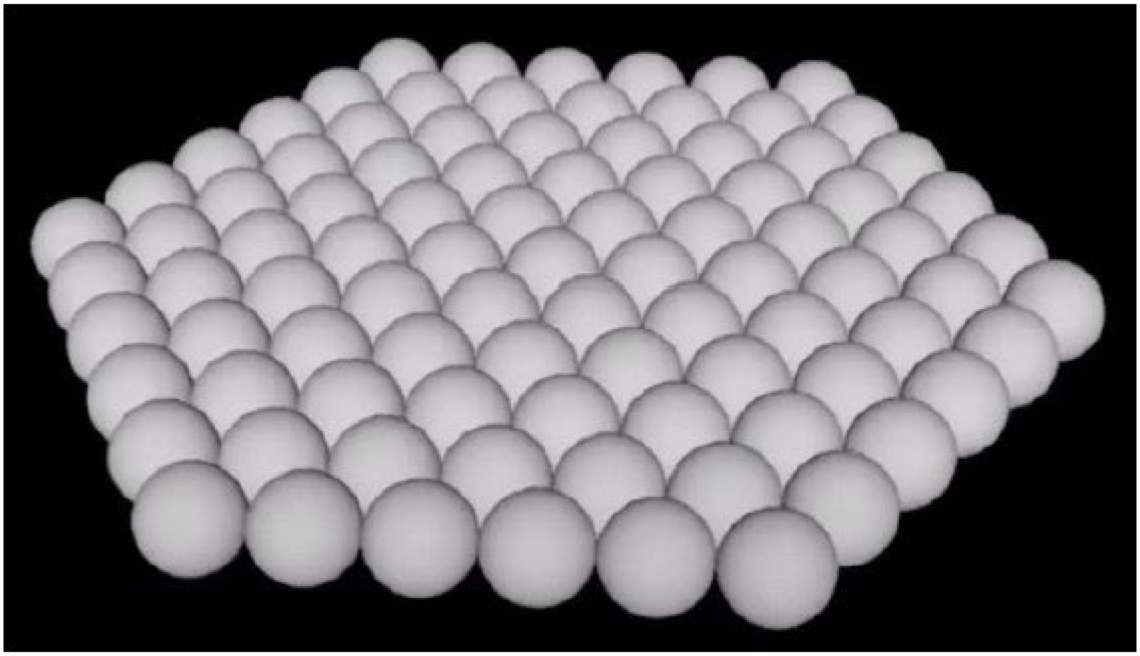


Figure 4-1: Regular Hexagonal Ball Arrangement

The ARC-100 core assembly has a duct inner wall distance of approximately 17.15 cm however the handling socket restricts the opening to about 10 cm in diameter at its smallest opening. The ball packing density and the total number of balls are calculated by ranging the ball diameter from 2.0 cm to 17.28 cm, equivalent to the duct inner wall distance. Using a 17.28 cm diameter ball is used to bound this study although it is not feasible to drop this size of a ball into an empty core assembly due to the ~10 cm opening in the core assembly handling socket. Table 4-1 shows the ball dimensions. The total number of balls and the ball packing density are plotted in Figure 4-2 and Figure 4-3, respectively. The 17.28 cm diameter balls would contact the duct inner wall surface, while the 15.28 cm diameter balls would allow a 1.0 cm gap from the duct inner wall surface. The 5.76 cm diameter balls would fit in two hexagonal rings with seven balls per layer. The 2.0 cm diameter ball would fit in four hexagonal rings with 37 balls.

As the ball diameter increases, the total number of balls decreases and the ball packing fraction increases. For the 2.0 cm diameter ball, the alternative shutdown system requires more than 3,000 balls, and the ball packing density is about 33%. For the 17.28 cm diameter ball, the alternative shutdown system requires nine balls, and the ball packing density is about 59%.

Table 4-1: Calculated Geometric Parameters of Ball Configurations

Ball Radius (cm)	1.00	2.88	7.64	8.64
Balls per layer	37	7	1	1
Number of Layers	89	29	10	9
Number of Balls	3,293	203	10	9
Ball Packing Density	0.33	0.49	0.45	0.59

[redacted]

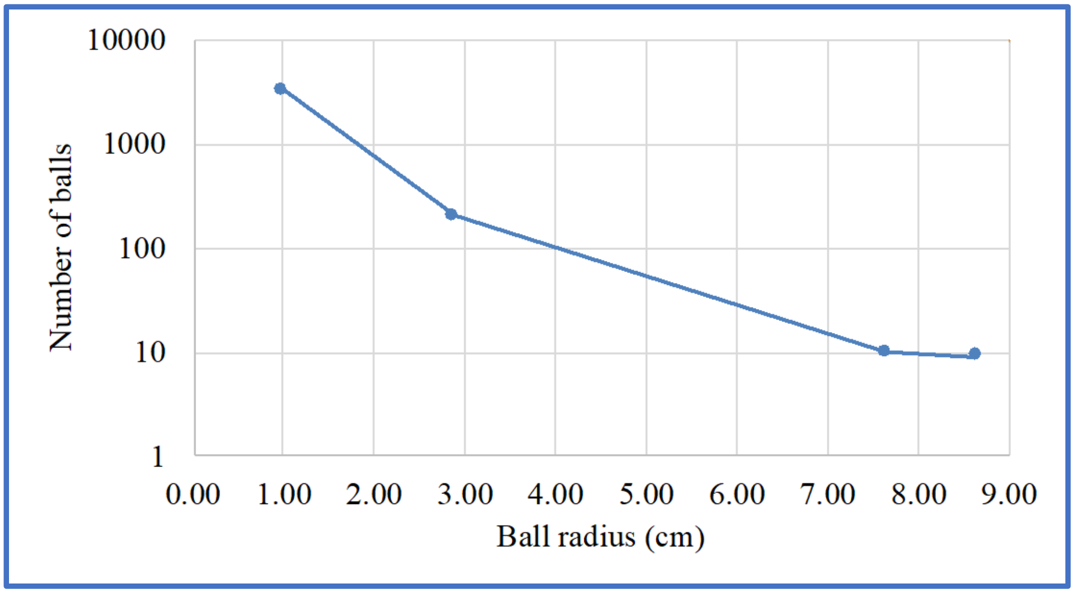


Figure 4-2: Number of Balls

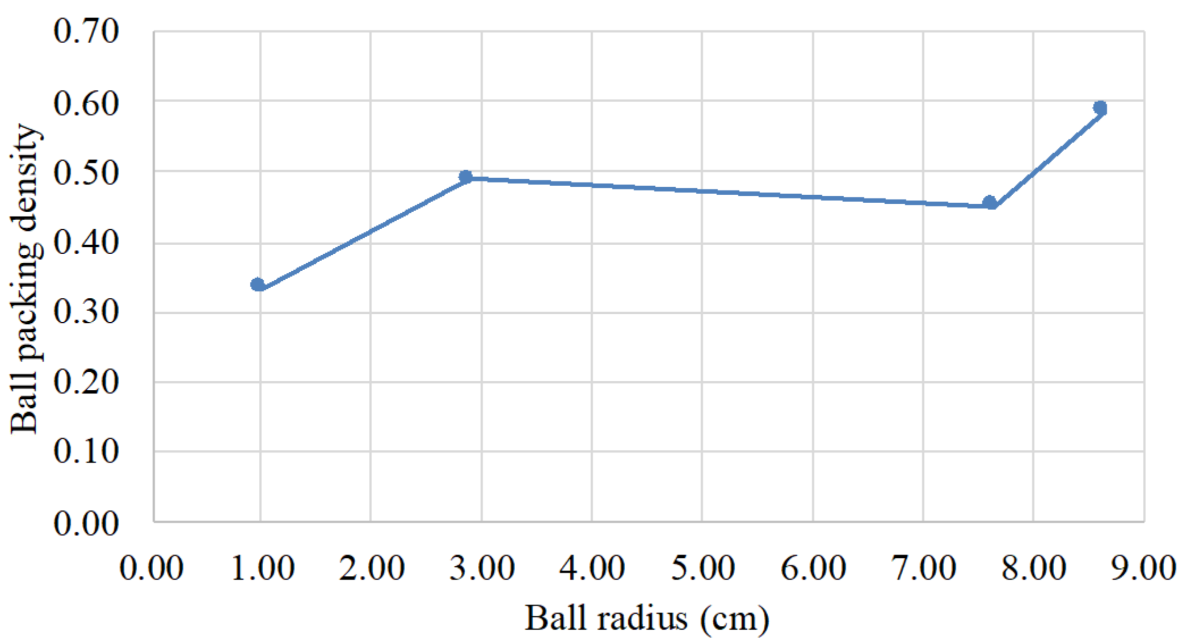


Figure 4-3: Ball Packing Densities

The material volume fractions for a given ball size are calculated based on the following assumptions. The balls were assumed to be clad in HT9. The ball cladding thickness in the Table 4-1 was intentionally selected to be greater than that of the fuel cladding, to minimize the dimension of the B_4C ball at its volume fraction. The thickness of the ball cladding can then be reduced thereby increasing the volume fraction of the B_4C, but at least initially it is desired to determine whether the B_4C can drive the reactor subcritical, so the packing fraction is determined conservatively. The B_4C smear density was assumed to be 90% for accommodating helium gas produced from neutron irradiation of boron. Whereas 85% smear density is taken in

[redacted]

[redacted] control rods, 90% smear density is assumed in the balls because they will sustain little neutron irradiation except for a short period when used in shutdown. Natural boron and enriched (45%) boron configuration have been assessed. In Figure 4-4, the B_4C volume fraction of the smaller balls is significantly lower than that of the larger balls, because the volume fraction of HT9 cladding is this case is significantly greater than that of the larger balls. Of course, as already stated the thickness of the ball cladding can be reduced, and the B_4C volume fraction be correspondingly increased.

[redacted]

Figure 4-4: Central Assembly Volume Fraction of Materials

Figure 4-5 shows the negative reactivity worth in terms of ball size and boron enrichment. The negative reactivity worth is roughly proportional to the B_4C volume fraction. The three large balls, having large negative reactivity worth, were also modeled using enriched boron. The enriched boron cases have about 500 pcm more negative reactivity than their natural boron counterparts.

The selection of ball size and boron enrichment can be made according to the required negative reactivity worth of the alternate shutdown system. The alternative shutdown system is required to bring the core subcritical from full-power operation to hot-standby condition by assuming that all primary control rods fail to scram, one control rod is accidentally fully withdrawn, and all secondary shutdown rods fail to scram. In the core design report [4], the temperature defect from the full power to hot standby is 0.22$, and the excess reactivity at BOL is 2.96$. Thus, the total required negative reactivity is 3.18$ or 2258 pcm (based on 710 pcm/$ at BOL). The result in Figure 4-5 shows that the B_4C balls with a diameter of 6.0 - 15.28 cm with enriched boron can provide the necessary negative reactivity, even if one were to assume all control rods are accidentally full withdrawn, but with little margin. However, it is assumed that only one of the six (having the highest worth) is withdrawn and the others remain at their highest location in the core, thereby providing a significant fraction of the required negative reactivity ($0.49 per rod).

So in reality the margin provided by theB_4C balls with the residual reactivity provided by the remaining control rods is sufficient to shut down the reactor.

The factor in Table 7.3 of Reference [4] that accounts for the ejection for the maximum worth rod, is already accounted above, reducing the required reactivity to 7.15$. The temperature defect from full power to hot standby to 0.22$ is provided inherently by the reactor, thereby reducing the require reactivity to 6.93$ If a large margin is required (1.80$ from Table 7.2 of Reference [4], then the boron carbide balls will not be sufficient to shut down the reactor, and another means will have to be identified. However (1) the worth of the carbide balls has been conservatively estimated (it can be higher by approximately 20% or more, i.e., 3.82$ instead of 3.18$ due to the higher packing fraction of 50% vs the assumed 40%) and (2) the quantity of B_4C present in the assembly could also be higher than assumed in the Reference [3] analysis.

The right ball size with boron enrichment should be determined that will fit through the handling socket of the ARC-100 core assembly. If other reactivity effects (such as shutdown margin of more than 1$) and uncertainties listed in the core design report [4] should be compensated, the total required reactivity is 7.62$ or 5410 pcm at BOL. The question is whether the alternative shutdown system coupled with the negative reactivity of the remaining 5 control rods should be able to compensate for such a high reactivity requirement if highly enriched boron is used.

Form Figure 4-5 and Appendix D, it can be seen that for the larger balls (such as the 39.5 mm radius ball analyzed in Appendix D) it is possible to achieve the required negative reactivity.

[redacted]

Figure 4-5: Negative Reactivity Worth of B_4C system (Export Controlled)

First it is noted that, as stated in Appendix D, the thickness of the cladding can be reduced to 0.5 mm from the 0.7 mm assumed in the Reference [3] analysis,[1] and the packing fraction can be greater than that calculated by Reference [3]. For balls of 39.5 mm OD, Appendix D

[1] The initial PRISM USS design proposed utilizing un-clad B_4C balls (Reference [12], page G.4.2-28).

[redacted]

[redacted] a packing fraction of 60%. Even with a packing fraction of 50% the reactivity would be increased from that stated in Reference [3] by about 20%. Second, the 1.80 factor in Table 2 of Reference [4] is a margin introduced for conservatism and could be reduced. The boron carbide balls should be able to shut down the reactor with an additional margin of about (7.15-3.81-2.96 =) 0.38$, instead of 1.8$.

In addition, the margins have been calculated based on an enrichment of Boron which is only 45%. It is possible to provide the balls with a significantly greater (although more expensive) enrichment. Enrichments up to 90% have been used. With such enrichment the worth of the B_4C balls would be approximately 1.7 times that with only 45% enrichment , or close to $ 6.0

[redacted]

5. Conclusions

5.1 Why an ARC-100 Alternate Shutdown System is Not Needed

Section 2 presented current ARC-100 means for reactivity control, which include 1) inherent reactivity feedback, 2) a primary (control) rod system, and 3) a secondary (safety) rod system.

Diversity has been incorporated into the design of the rod systems, as described in Section 2.2. For example, the six (6) primary (control) rods have been divided into two (2) groups, with the drivers for each group being procured from a different manufacturer. Likewise, the solenoids for each group will be procured from different manufacturers. Although for the analyses presented in this white paper the secondary (safety) rod system drivers and solenoids are from the same manufacturers as one of the primary (control) rod system groups, this is primarily due to the scarcity of manufacturers. If three (or more) manufacturers can be identified, then to add diversity the secondary (safety) rod system drivers and solenoids would be procured from a manufacturer independent of the two primary (control) rod system manufacturers.

Additionally, insertion of the primary (control) rods following a SCRAM is by gravity alone, whereas insertion of the secondary (safety) rods following a SCRAM is by spring assisted gravity. Also, during operation the secondary (safety) rods are positioned above the active core.

Section 2.3 summarize the results of a probabilistic safety analysis (PSA) for the ARC-100 Reactor Protection System (RPS), which showed that with the assumption that the primary (control) rod with the largest worth withdrawn or ejected from the core, the RPS failure probability was calculated to be 1.371×10^{-5} demand^{-1} (not including the probability of rod withdrawal / ejection, which for EBR-II was estimated to be 3.3×10^{-2} yr^{-1} for a combined probability of 4.52×10^{-7}); and with all six primary (control) rods available, the RPS failure probability was calculated to be 6.175×10^{-8} demand^{-1}. The PSA demonstrates the reliability of the current ARC-100 RPS to shut down the reactor.

Should the RPS fail to shut down the reactor, the ARC-100's inherent reactivity feedback controls the fission process within the core, as described in Section 2.1. The ARC-100's inherent reactivity feedback will self-regulate itself to a power level equal to the amount of heat being transferred out, without control or safety rod movement. The ARC-100 can remain in a controlled state (but remain critical) for an indefinitely long period, without any fuel clad breaching. To transition from the controlled state to safe shutdown (as required by SFR DC 26(1)) the primary (control) and/or secondary (safety) rods must be inserted. This would be performed by the operator manually SCRAMing the reactor.

Manually SCRAMing the reactor would cause all nine (9) primary (control) and secondary (safety) rods to be inserted, unless core deformation presents the rods from inserting.

As described in Section 2.4, ANL has performed an analysis that determined that the maximum deformation of the fuel assemblies to be between 0.29 and 0.31 cm at BOL and EOL respectively, with the control rods experiencing lesser deformation (Reference [19]). ANL concluded that it is highly unlikely that the core would deform sufficiently to cause a binding of the control rods. Nonetheless, to provide further assurance that core deformations would not prevent rod insertions, the pull-out force for assemblies, particularly those near the control and safety rods, will be periodically measured, and compared to the forces predicted by Reference [19].

[redacted]

Thus, for the reasons described above and the more detailed analyses presented in Section 2, and the appendices and references, it is concluded that the current design of the ARC-100 reactivity control means comply with SFR DC 26 and an Alternative Shutdown System is not needed. ARC Clean Technology seeks the Nuclear Regulatory Commission's opinion on this conclusion of the current ARC-100 means for reactivity control.

5.2 If Needed, a Proposed ARC-100 Alternate Shutdown System

Disregarding the Section 5.1 conclusion, a conceptual design of an ARC-100 Alternative Shutdown System has been developed and is described in Section 4 and Reference [3]. In this design highly enriched boron carbide (B_4C) balls drop by gravity into the currently empty center position of the ARC-100 core (Figure 2-2). This design is similar to the initial proposed PRISM Ultimate Shutdown System design (Appendix C.1.2), which received a positive review by NRC in the PRISM Preapplication Safety Evaluation Report (Reference [13]).

Should the NRC's opinion on the current ARC-100 means for reactivity control (requested above) not be positive, then ARC Clean Technology seeks the Nuclear Regulatory Commission's opinion on the acceptability of this Alternative Shutdown System design.

[redacted]

6. References

[1] "Passive Shutdown Systems for Fast Neutron Reactors" IAEA Nuclear Energy Series No. NR-T-1.16, 2020

[2] "Absorbers, material, control rods and design for shutdown system for advance liquid metal fast reactors', IAEA-TECDOC-884, June 1996

[3] A. Kassam, N. Stauff, E. Hoffman, T.K. Kim, "Alternate Shutdown System for the ARC-100", ANL-NSE-ARC/ARC29-03, April 21, 2022

[4] E. Hoffman, T. Fei, T. Kim, "286 MWth ARC-100 Core Design Report – Rev. 06," Argonne National Laboratory, June 22, 2023.

[5] Weisstein, Eric W. "Sphere Packing." MathWorld--A Wolfram Web Resource. https://mathworld.wolfram.com/SpherePacking.html

[6] Ethridge, J.L., Pacific Northwest Laboratory (PNL) and R.B. Baker, R.D. Leggett, A.L. Pitner, and A.E. Waltar, Westinghouse Hanford Company (WHC), "Fast Flux Test Facility Core System," WHC-SA-0981, November 1990.

[7] Pitner, A.L., "Liquid Metal Reactor Absorber Technology," Westinghouse Hanford Company (WHC), WHC-SA-0976, October 1990.

[8] U.S. Nuclear Regulatory Commission (NRC), "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Regulatory Guide 1.232, Revision 0, ADAMS No.: ML17325A611, April 2018.

[9] U.S. Nuclear Regulatory Commission (NRC), "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," SECY-94-084, ADAMS No.: ML003708068, March 28, 1994.

[10] TerraPower, LLC., "Principal Design Criteria for the Natrium Advanced Reactor," Topical Report TP-LIC-LET-0128, Revision 1, ADAMS No.: ML24101A362, April 10, 2024.

[11] U.S. Nuclear Regulatory Commission (NRC), "Terrapower, LLC – Draft Safety Evaluation of Topical Report NATD-LIC-RPRT-0002, "Principal Design Criteria for the Natrium Advanced Reactor," Revision 1 (EPID L-2021-TOP-0020)," ADAMS No.: ML24103A212, April 12, 2024.

[12] GE Nuclear Energy, "Power Reactor Innovative Small Module, Preliminary Safety Information Document, Volume VI, Appendix G Responses to Issues in DSER," GEFR-00793, March 1990.

[13] U.S. Nuclear Regulatory Commission (NRC), "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NUREG-1368, ADAMS No.: ML063410561, February 1994.

[14] B.S. Triplett, E.P. Loewen, And B.J. Dooies, "PRISM: A Competitive Small Modular Sodium-Cooled Reactor," GE Hitachi Nuclear Energy, Nuclear Technology, Vol. 178, May 2012.

[redacted]

[15] ARC Clean Technology, LLC., "ARC-100 Plant, Reactor Protection System Description and Failure Probability Modeling," Technical Report ARC20-RPS-001, Revision 0, December 2023.

[16] Ma, Zhegang, Thomas E. Wierman, and Kellie J. Kvarfordt, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants: 2020 Update," Idaho National Laboratory (INL), INL/EXT-21-65055, November 2021.

[17] Smith, C.L. and S.T. Wood, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8," Idaho National Laboratory (INL), NUREG/CR-7039, Vol. 1 to 7, June 2011.

[18] Idaho National Laboratory (INL), "CCF Parameter Estimates Spreadsheet 2020," Rev. 1, https://nrcoe.inl.gov/ccf_pe/

[19] N. Stauff, K. Ramey, M.A. Smith, N. Wozniak, T. Sumner, T.K. Kim, "Final Report on Predictive Analysis of PRD as Function of Deformation and Other Anomalies," ANL/NSE-ARC/ARC20 -23, JUNE 30, 2024.

[20] U.S. Nuclear Regulatory Commission (NRC), "Safety Evaluation; Related to the Kairos Power LLC Construction Permit Application for the Hermes Test Reactor," Docket 50-7513, ADAMS No.: ML23158A268, June 2023.

[21] Argonne National Laboratory, Experimental Breeder Reactor II(EBR II), Systems Design Descriptions, Chapter 2 Section 2.7, 15 February 1985

[22] T.A. Pitterle, and H.O Lagally, "Review of FFTF and CRBRP Control Rod Systems Designs, Conference 7712170-5, October 4, 1977

APPENDIX A: SFR DC 26 NRC RATIONALE FOR ADAPTION

Recent licensing activity, associated with the application of GDC 26 and GDC 27 to new reactor designs (ADAMS Accession Nos. ML16116A083 (Ref. 29) and ML16292A589) (Ref. 30), revealed that additional clarity could be provided in the area of reactivity control requirements. ARDC 26 combines the scope of GDC 26 and GDC 27. The development of ARDC 26 is informed by the proposed general design criteria of 1965 (AEC-R 2/49, November 5), 1967 (32 FR 10216) (Ref. 31), current GDC 26 and 27, the definition of safety-related SSC in 10 CFR 50.2, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs” (Ref. 32), and the prior application of reactivity control requirements.

(1) Currently the second sentence of GDC 26 states, that one of the reactivity control systems shall use control rods and shall be capable of reliably controlling reactivity changes to ensure that, under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The staff recognizes that specifying control rods may not be suitable for advanced reactors. Additionally, reliably controlling reactivity, as applied to GDC 26, has been interpreted as ensuring the control rods are capable of rapidly (i.e., within a few seconds) shutting down the reactor (ADAMS Accession No. ML16292A589) (Ref. 30).

The staff changed control rods to “means” in recognition that advanced reactor designs may not rely on control rods to rapidly shut down the reactor (e.g., alternative system designs or inherent feedback mechanisms may be relied upon to perform this function). The wording of “reliably controlling reactivity” in GDC 26 has been replaced with “inserting negative reactivity at a sufficient rate and amount” to more clearly define the requirement. For a non-LWR design the rate of negative reactivity insertion may not necessitate rapid (within seconds) insertion but should occur in a time frame such that the fission product barrier design limits are not exceeded.

The term “specified acceptable fuel design limits” is replaced with “design limits for fission product barriers” to be consistent with the AOO acceptance criteria while also addressing liquid fueled reactors which may not have SAFDLs. ARDC 10 and ARDC 15 provide the appropriate design limits for the fuel and reactor coolant boundary, respectively.

The wording “safe shutdown is achieved and maintained…” has been added again to more clearly define the requirements associated with reliably controlling reactivity in GDC 26. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs” (Ref. 32), describes the characteristics of a safe shutdown condition as reactor subcriticality, decay heat removal, and radioactive materials containment. ARDC 26 (1) clearly defines that reactor shutdown at any time during the transient is the performance requirement. The distinction between during and following the transient is discussed in (2) below.

In regard to safety class, the capability to insert negative reactivity at a rate and amount to preserve the fission product barrier(s) and to shut down the reactor during an AOO is

[redacted]

[redacted] as a function performed by safety-related SSCs in the 10 CFR 50.2 definition of safety-related SSCs.

(2) The first sentence of GDC 26, states that two independent reactivity control systems of different design principles shall be provided. The third sentence of GDC 26, states that the second reactivity control system shall be capable of reliably controlling the rate of changes resulting from planned, normal power changes (including xenon burnout) to assure specified acceptable fuel design limits are not exceeded. ARDC 26 (2) is consistent with the current requirements of the second reactivity control system specified in GDC 26. The words “including xenon burnout” have been deleted as this may not be as important for some non-LWR reactor designs. Also, “of different design principles” from the first sentence of GDC 26 has been replaced with “independent and diverse” to clarify the requirement. The reactivity means given by ARDC 26 (2) is a system important to safety but not necessarily safety-related as it does not mitigate an AOO or accident but is used to control planned, normal reactivity changes such that the design limits for the fission product barriers are preserved thereby minimizing challenges to the safety-related reactivity control means or protection system.

The term “independent and diverse” indicates no shared systems or components and a design which is different enough such that no common failure modes exist between the system or means in ARDC 26 (2) and safety-related systems in ARDC 26 (1) and (3).

(3) Current GDC 27 states that the reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. Reliably controlling reactivity, as applied to GDC 27 requires that the reactor achieve and maintain a safe, stable condition, including subcriticality, using only safety related equipment with margin for stuck rods (ADAMS Accession No. ML16116A083) (Ref. 29).

ARDC 26 (3) is written to clarify that shut down following a postulated accident using safety-related equipment or means is required. The term “following a postulated accident” refers to the time when plant parameters are relatively stable, no additional means of mitigation are needed and margins to acceptance criteria are constant or increasing. ARDC 26 allows for a return to power during a postulated accident consistent with the current licensing basis of some existing PWRs if sufficient heat removal capability exists (e.g., PWR main steam line break accident), but ARDC 26 (1) precludes a return to power during an AOO.

(4) The fourth sentence of GDC 26 regarding the capability to reach cold shutdown has been generalized in ARDC 26 (4) to refer to activities which are performed at conditions below (less limiting than) those normally associated with safe shutdown. SECY-94-084 (Ref. 32) describes staff positions on obtaining a cold shutdown and explains that the requirement to bring the plant to cold shutdown is driven by the need to inspect and repair a plant following an accident. In regard to safety class, the capability to bring the plant to a cold shutdown is not covered by the definition of safety-related SSCs in 10 CFR 50.2, and most operating pressurized-water reactors have not credited safety-related SSCs to satisfy this requirement of GDC 26. Based on the information provided above, the system credited for holding the reactor subcritical under conditions necessary

[redacted]

[redacted] activities such as refueling, inspection and repair is identified as an important to safety system.

A.1.1 RG 1.232, Appendix B, SFR DC 26 References

29. U.S. Nuclear Regulatory Commission, “Response to Gap Analysis Summary Report for Reactor System Issues,” September 2016. (ADAMS Accession No. ML16116A083).

30. U.S. Nuclear Regulatory Commission, “Response to NuScale Gap Analysis Summary Report for Reactivity Control Systems, Addressing Gap 11, General Design Criteria 26,” December 2016. (ADAMS Accession No. ML16292A589).

31. U.S. Atomic Energy Commission, “Proposed General Design Criteria of 1965, AEC-R 2/49,” November 5, 1967. (32 FR 10216).

32. U.S. Nuclear Regulatory Commission, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 1994. (ADAMS Accession No. ML003708068).

APPENDIX B: ARC-100 RPS COMMON CAUSE FAILURES

B.1 ARC-100 Reactor Protection System (RPS)

The ARC-100 Reactor Protection System (RPS) inserts the control rods into the core to ensure fuel integrity and to protect the reactor coolant boundary under abnormal operating conditions. The RPS does this by monitoring reactor parameters and comparing their values to setpoints. If a setpoint is exceeded, then the RPS issues a trip signal. If two or more trip signals are detected, then the RPS will scram the reactor. See Section B.1.1 for a more complete description of the RPS scram logic. This section focuses on describing the physical configuration of the RPS.

The ARC-100 RPS consists of two reactivity control systems: a Primary RPS that controls the primary control rods and is designated as RPS1 and a Secondary RPS that controls the secondary safety rods and is designated as RPS2. Both systems are classified as Safety Related.

Figure B-1 presents a schematic of the RPS, showing both RPS1 and RPS2. In Figure B-1 RPS1 is shown on the left, while RPS2 is shown on the right. Additionally, primary control rods A, B, and C constitute RPS1, Group 1 and rods E, F, and G constitute RPS1, Group 2.

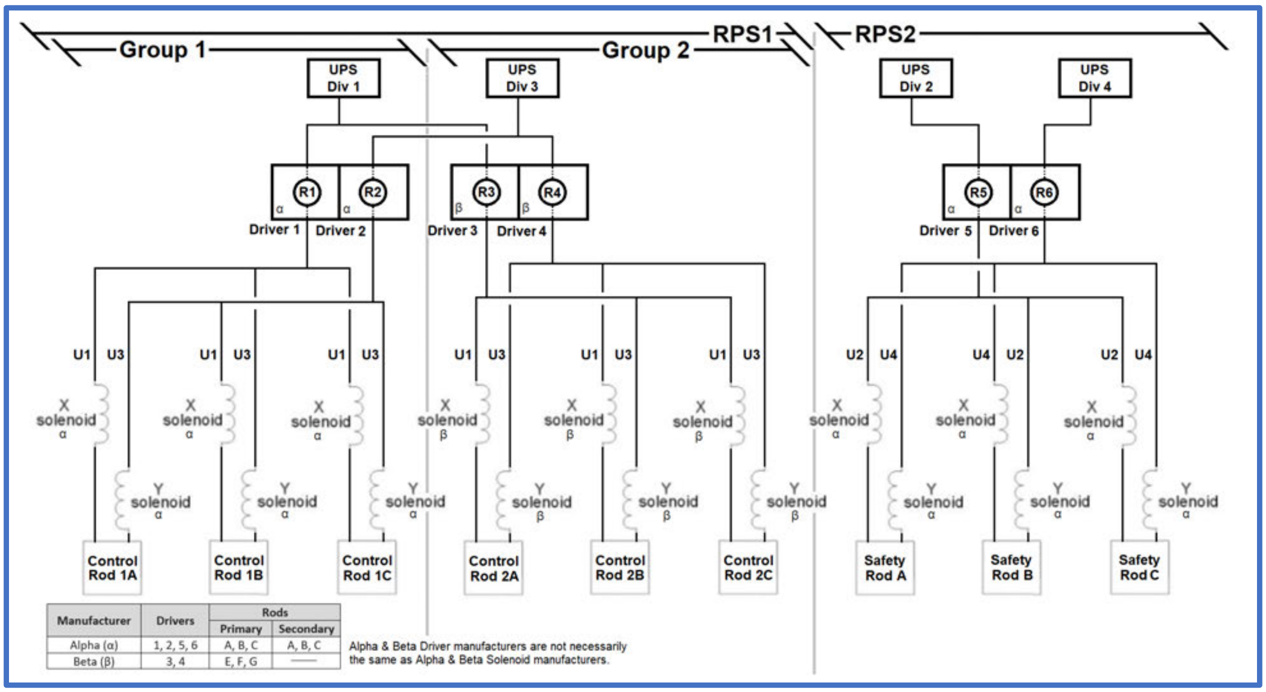


Figure B-1: Reactor Protection System Schematic

As shown, the RPS is powered by the ARC-100 four (4) division uninterruptible power supplies (UPS)—two (2) UPS divisions for RPS1 and two (2) UPS divisions for RPS2. Power is fed to the load drivers, which in turn feed power to the solenoids. When the solenoids are powered, or energized, they hold the rod in place. Upon receiving a scram signal the load driver turns off power to the solenoids, causing them to open, and allowing the rod to drop by gravity into the core. Each rod has two (2) solenoids and both solenoids must be de-energized before the rod

[redacted]

drops. This design allows on-line testing of one solenoid while the other solenoid prevents the rod from dropping into the core.

To introduce diversity into the RPS it is specified that the load drivers and solenoids will each be obtained from at least two (2) different manufacturers. In Figure B-1 the two manufacturers are identified as alpha and beta (note, the alpha and beta load driver manufacturers are not necessary the same as the alpha and beta solenoid manufacturers). Specifying different manufacturers will reduce (but not eliminate) the probability of a common cause failure (see Section B.2.1).

As stated above, the RPS monitors reactor parameters to determine if a trip signal is warranted. The preliminary list of RPS1 and RPS2 monitored parameters is provided in Table B-1.

Table B-1: RPS Trip Parameters

RPS	Parameter	RUN Mode	STARTUP Mode
RPS1 Primary	Absolute flux — high	ü	ü
	Flux to flow ratio — high	ü	—
	Core inlet temperature — high	ü	ü
	Core exit temperature — high	ü	ü
	More than one EM pump VFD set breaker — open	ü	—
	Short Power Period	—	ü
RPS2 Secondary	RPS1 has sent a scram signal	ü	ü
	Core inlet temperature — high-high	ü	ü
	Core exit temperature — high-high	ü	ü

The RPS2 core inlet and exit temperature setpoints are set higher than the RPS1 temperature setpoints.

Finally, both the RPS1 and RPS2 include a manual SCRAM that directly removes power from the load drivers, causing the solenoids to open, and the rods to insert. No software is involved in this SCRAM, and it is always available in both the main control room and at the alternate control panel. Credit for this manual SCRAM feature has not been taken in this analysis.

B.1.1 RPS Scram Logic

The RPS decision logic comprises a two out of four (2/4) logic matrix that consists of four independent channels. The RPS scram logic is shown diagrammatically in Figure B-2, and is described below.

Each channel compares each of the monitored parameters to their setpoint, and if the setpoint is exceeded it sends a signal to the RPS2 equivalent channel that it might have to scram. For example, RPS1 Channel 1 would send that signal to the RPS2 Channel 1.

If any two or more RPS1 channels receive signals that their setpoints have been exceeded (either for the same parameter or different parameters), then using the 2/4 logic of RPS1 the reactor primary control rods are scrammed. Independently, if RPS2 receives two (2) or more “might have to scram” signals from RPS1 it will scram the secondary safety rods.

[redacted]

This does not mean that the scram of the primary control rods sends a signal to scram the secondary safety rods, but the effect of the channels of RPS1 sending signals to their equivalent channels of RPS2 sets the condition that if the RPS1 logic satisfies the 2/4 requirement and scrams the primary control rods, the 2/4 logic of RPS2 will also be satisfied and the secondary safety rods will also scram.

However, if the RPS1 channel fails to send a signal to its corresponding RPS 2 channel it is possible that RPS2 will not have sufficient signals to satisfy its 2/4 logic. So, the primary control rods would be scrammed, but the secondary safety rods might not scram, unless they do so based on the 2/4 logic of its own channels receiving signals that the high-high temperature is exceeded. The secondary safety rod scram in this instance would be a bit later than the primary control rod scram.

Finally, if RPS1 2/4 logic is not satisfied, RPS2 2/4 logic can still be satisfied if those channels receiving inputs from the high -high temperature have at least 2 channels registering a temperature higher than the setpoint. In this case the primary control rods are not scrammed, but the secondary safety rods are scrammed.

[redacted]

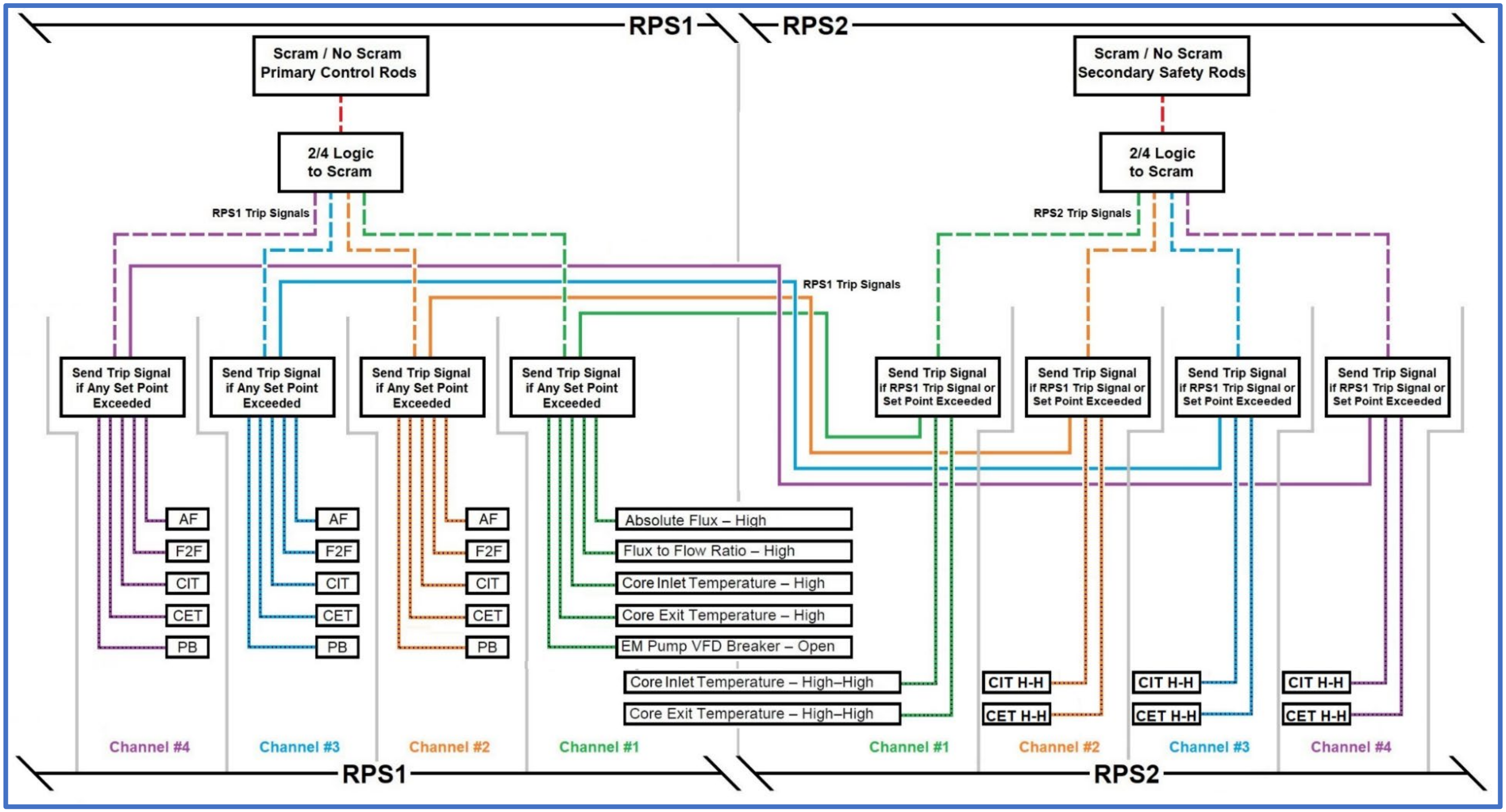


Figure B-2: Reactor Protection System (RPS) Scram Logic

B.2 Common Cause Failures

Common cause failures (CCFs) can be defined as the failure of two or more components, subsystems, or systems due to a shared cause or mechanism, which has not been accounted for explicitly. CCFs are important since they defeat redundancy and/or diversity, and often have a high probability of occurrence relative to the combination of random independent failures of components, subsystems or systems.

CCF is modeled realistically by using alpha factors and adjusting these factors to reflect load drivers and solenoids' diversity. Although the failure modeling approach follows realism, it is only to the extent of what is practicable, following NRC's policy statement on the use of PRA methods [1]. This principle was applied by modeling a CCF in an aggregated basic event rather than modeling all possible permutations explicitly because the latter would create too much complexity in the fault trees. For example, a CCF of 5 components in a common-cause component group (CCCG) size 6 requires 347 permutations, and a CCF of 6 components in a group of 6 requires 202 permutations. It is impractical to model these permutations explicitly in the fault tree.

A number of models have been developed that can be used to estimate the probability of a common-cause event involving a specific number of components in a CCCG of a specific size, including the Basic Parameter model, the Beta model, the Multiple Greek Letter (MGL) model, and the Alpha Factor model. Both the MGL and Alpha models are programmed into SAPHIRE's CCF calculator [2]. This RPS failure analysis modeled CCF using the Alpha model option. Two (2) types of alpha factors exist: 1) "demand" when a "per demand" probability is utilized and 2) "rate" when a "per year" probability is utilized. An example utilizing "demand" alpha factor is given Figure B-3.

The SAPHIRE CCF calculator performs the events' permutations and sums the total probability as a basic event's probability. It makes the fault trees more readable to human analysts. However, there is a downside that it may overestimate CCF probabilities in certain cases. Such overestimations are deemed acceptable because: (1) they err towards conservatism, and (2) there is little or a diminishing return in modeling hundreds of CCF combinations explicitly; it is estimated that the RPS failure probability when a rod is ejected decreases only slightly by less than 6% (8×10^{-8}) when CCFs are calculated in a more stringent detail.

[redacted]

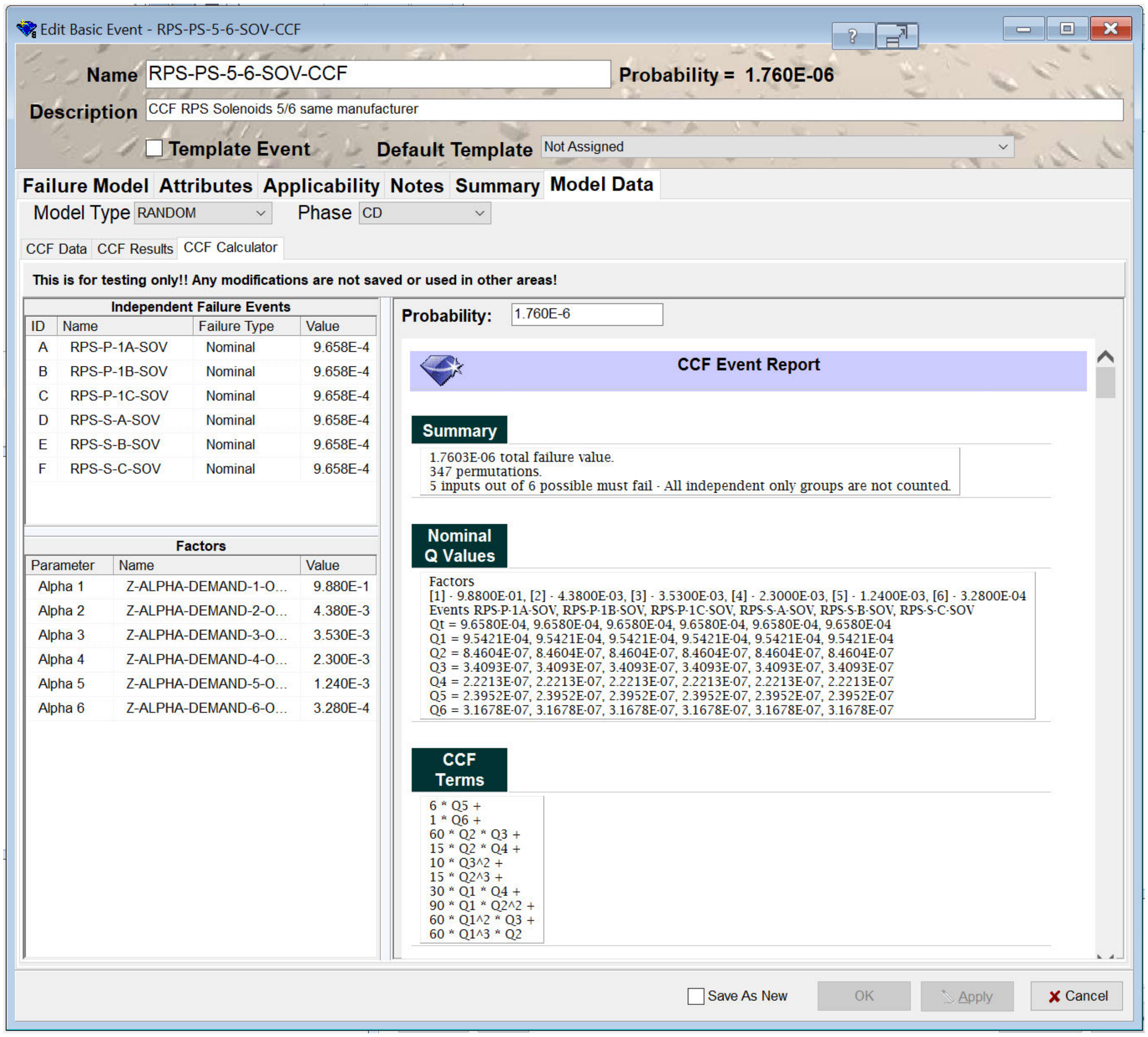


Figure B-3: Example SAPHIRE CCF Probability Calculation

B.2.1 Effect of Diverse Manufacturers

There are numerous specific individual failures that can cause a CCF. However, because CCF events are scarce, INL/EXT-21-33376 [5] suggests that the following five cause groups be used instead of specific individual failures: Component, Design, Environment, Human, and Other. Although the cause group approach may miss some details of specific failure causes, it is sufficient to not lose important information while, at the same time, it represents a manageable set that can be tracked and analyzed, and is supported by the CCF data [3]. Table 3-1 lists the five cause groups and provides descriptions of the types of specific individual failures that are included in each group.

Table B-2: Component Failure Causes Within the Five Cause Groups

Cause Group	Description
Component	A failure results from a mechanism internal to the component other than aging or wear; this applies to erosion/corrosion, equipment fatigue, and internal contamination.
	A failure results from setpoint drift or adjustment.
	A failure results from a failure internal to the component that failed due to aging or wear.
Design	A construction or installation error was made during the original or modification installation (e.g., installation of an incorrect component or material, or specifying an incorrect component or material).
	A design error was made.
	A manufacturing error was made when manufacturing the component.
Environment	A failure results from an environmental condition at the component location; this applies to: Chemical reactions; Electromagnetic interference; Fire/smoke; Impact loads; Acts of nature; Radiation (irradiation); Temperature (abnormally high or low); Vibration loads (excluding seismic events).
	A failure results from an environmental condition that is transitory in nature and places a higher-than-expected load on the equipment.
	A failure results from an internal environment condition; this applies to: Debris; Foreign material; Operating medium chemistry issue.
Human	A human error incurred when performing an operational activity results in an unintentional or undesired action.
	A human error incurred when performing a maintenance activity results in an unintentional or undesired action.
	The correct procedure was not followed; this applies to: Calibration/test staff; Construction/installation/modification staff; Maintenance staff; Operations staff; Other plant staff.
	A failure results from an inadequate procedure; this applies to: Calibration/test procedures; Construction/installation/modification procedures; Maintenance procedures; Operational procedures; Administrative procedures; Other procedures.
Other	A failure results from a component state not associated with the component that failed. One example would be he diesel generator failed due to having no fuel in the fuel storage tanks.
	The failure cause is provided but does not meet any one of these descriptions.
	The failure cause is not known.

Source: INL/RPT-23-72728 [3], Table 1-1

Each of the five cause groups contributes to the overall CCF alpha factor.

Table B-3 shows the contributions of each of the five cause groups to the “demand” alpha factors for CCCG of 2 through 6. A similar table exists for the five cause groups contributions to the “rate” (i.e., per year) alpha factors.

Table B-3: CCF Cause Group Alpha Factor Contributors — Demand

Failures	Component	Design	Environment	Human	Other
1 of 2	3.79E-01	1.68E-01	2.95E-02	2.92E-01	1.18E-01
2 of 2	1.73E-03	5.24E-03	4.95E-04	5.57E-03	3.07E-04
1 of 3	3.81E-01	1.67E-01	2.92E-02	2.92E-01	1.18E-01
2 of 3	2.05E-03	3.44E-03	6.33E-04	3.56E-03	2.70E-04
3 of 3	4.21E-04	1.86E-03	1.62E-04	1.79E-03	9.83E-05
1 of 4	3.80E-01	1.66E-01	2.90E-02	2.92E-01	1.19E-01
2 of 4	2.31E-03	2.53E-03	7.68E-04	2.50E-03	3.32E-04
3 of 4	5.02E-04	1.74E-03	1.54E-04	1.62E-03	4.47E-05
4 of 4	2.06E-04	7.18E-04	9.09E-05	6.42E-04	4.83E-05
1 of 5	3.81E-01	1.65E-01	2.90E-02	2.93E-01	1.19E-01
2 of 5	1.68E-03	1.87E-03	5.43E-04	1.49E-03	2.90E-04
3 of 5	7.76E-04	1.46E-03	3.51E-04	1.35E-03	9.04E-05
4 of 5	2.97E-04	9.16E-04	1.07E-04	8.79E-04	2.13E-05
5 of 5	8.76E-05	2.99E-04	1.45E-05	2.70E-04	3.26E-06
1 of 6	3.81E-01	1.66E-01	2.91E-02	2.93E-01	1.19E-01
2 of 6	1.32E-03	1.39E-03	3.21E-04	1.03E-03	2.86E-04
3 of 6	7.53E-04	1.29E-03	3.60E-04	1.02E-03	8.62E-05
4 of 6	3.77E-04	8.55E-04	1.61E-04	8.55E-04	3.67E-05
5 of 6	1.74E-04	5.24E-04	5.10E-05	4.75E-04	9.01E-06
6 of 6	5.80E-05	1.31E-04	1.42E-05	1.12E-04	1.06E-05

The sum over all failures and all five cause groups of the alpha factors within a CCCG is always one (1). If one or more cause group does not apply to a particular CCF, then the sum of all the alpha factors will be less than one (1), and the calculated CCF probability will be smaller than if all five cause groups were present.

As stated in Section B.1.1, to introduce diversity into the RPS it is specified that the load drivers and solenoids will each be obtained from at least two (2) different manufacturers. When different manufacturers are involved the Component and Design cause groups do not apply to the CCF alpha factors.

Table B-4 lists the CCF alpha factors that were used in this analysis. The CDEHO, or full, factors were obtained from "CCF Parameter Estimates Spreadsheet 2020, Rev. 1" [4], and used whenever all the components of a common-cause component group (CCCG) are from the same manufacturer. The EHO factors were calculated as described above and used whenever the components of a CCCG are from two (2) or more manufacturers. Comparing the EHO columns

[redacted]

to the CDEHO columns shows that using different manufacturers reduces the overall CCF alpha factors by a little over 50%.

Table B-4: CCF Alpha Factors

CCCG	Alpha	Failures	Demand		Rate	
			CDEHO	EHO	CDEHO	EHO
CCCG 2	α1	1 of 2	9.87E-01	4.40E-01	9.78E-01	4.73E-01
	α2	2 of 2	1.35E-02	6.37E-03	2.17E-02	1.69E-02
CCCG 3	α1	1 of 3	9.86E-01	4.38E-01	9.79E-01	4.72E-01
	α2	2 of 3	1.00E-02	4.47E-03	1.26E-02	7.84E-03
	α3	3 of 3	4.37E-03	2.05E-03	8.28E-03	7.19E-03
CCCG 4	α1	1 of 4	9.86E-01	4.40E-01	9.81E-01	4.71E-01
	α2	2 of 4	8.52E-03	3.60E-03	9.68E-03	6.36E-03
	α3	3 of 4	4.10E-03	1.82E-03	6.20E-03	4.35E-03
	α4	4 of 4	1.72E-03	7.81E-04	3.53E-03	3.03E-03
CCCG 5	α1	1 of 5	9.87E-01	4.41E-01	9.83E-01	4.72E-01
	α2	2 of 5	5.92E-03	2.32E-03	6.58E-03	4.33E-03
	α3	3 of 5	4.06E-03	1.79E-03	5.39E-03	3.45E-03
	α4	4 of 5	2.24E-03	1.01E-03	3.71E-03	2.79E-03
	α5	5 of 5	6.82E-04	2.88E-04	1.36E-03	1.17E-03
CCCG 6	α1	1 of 6	9.88E-01	4.41E-01	9.85E-01	4.40E-01
	α2	2 of 6	4.38E-03	1.64E-03	4.56E-03	1.94E-03
	α3	3 of 6	3.53E-03	1.47E-03	4.66E-03	1.65E-03
	α4	4 of 6	2.30E-03	1.05E-03	3.32E-03	1.31E-03
	α5	5 of 6	1.24E-03	5.35E-04	2.14E-03	8.72E-04
	α6	6 of 6	3.28E-04	1.37E-04	5.92E-04	2.57E-04

[redacted]

APPENDIX C: SHUTDOWN SYSTEMS (INCLUDING ALTERNATE) PROPOSED BY OTHER REACTORS

This appendix contains a description of alternate shutdown system at three advanced reactor designs: 1) Power Reactor Innovative Small Module (PRISM), 2) TerraPower's Natrium Advanced Reactor, and 3) Hermes Non-Power Reactor.

C.1 PRISM Ultimate Shutdown System (USS)

There are two iterations of the PRISM design—the initial 1987 design and a revised 2006 design.

C.1.1 Initial USS

A Power Reactor Innovative Small Module (PRISM) conceptual design Preliminary Safety Information Document (PSID) was submitted to the NRC by General Electric on behalf of the U.S. Department of Energy in December 1987. PRISM, Appendix G (March 1990) describes significant design changes that were made in response to concerns raised by the NRC staff, including the addition of an Ultimate Shutdown System (USS) (Reference [12]).

Below is a slightly edited description of the PRISM USS from the PSID, Appendix G.

> ***G.4.2.2.5 Ultimate Shutdown System***
>
> *The ultimate shutdown system (USS) provides a diverse, independent means of bringing the reactor to cold shutdown. During an anticipated transient without scram event (…) strong inherent negative reactivity feedback with rising temperature maintains the reactor at a stable but hot equilibrium state. Final cold shutdown can be achieved with the USS by operator action to release neutron absorbing balls containing fully enriched boron-10 from a container at the closure head which fall by gravity into an open assembly in the center of the core.*
>
> *For the control rods to fail to insert into the core, severe distortion of the core internals must occur. The B4C balls are relatively small in size and have the characteristic of being able to follow a torturous path. So even under distorted conditions, it is reasonable to conclude that the balls will be able to move into the center core assembly and accomplish the cold shutdown function.*
>
> *The USS is illustrated in* [Figure C-, while Figure C-2 shows its location within the core]. *B_4C balls are stored in a dry canister within the reactor above the sodium. Upon actuation, the balls fall by gravity down a guide tube into an open thimble at the core center. The worth of absorber inserted into the core is sufficient to bring the reactor from 135% of full power to a cold shutdown.*
>
> *The system is made up of four subassemblies which include the absorber storage canister, center shutdown absorber assembly, absorber guide tube, and guide tube drive mechanism. These subassemblies are described in the paragraphs following.*
>
> *The absorber storage canister maintains the absorber material in a dry condition and out of the core position during normal reactor operating and shutdown periods. The absorber consists of approximately one cubic foot of 0.25-inch*

[redacted] fully enriched B_4C balls. The balls may be clad in a metallic jacket if testing shows a problem with the B_4C cracking.

The balls are retained in the canister by a hinged, sealed door secured by pull pins on each side of the hinge. … The hinged bottom folds down allowing the balls to enter the absorber guide tube. …

The guide tube extends from the closure to the core and channels the B4C balls from the absorber storage canister to the center core assembly. …

The center core assembly maintains the B4C absorber in the proper location once the ultimate shutdown system has been actuated. … During reactor operation the center assembly is a sodium-filled hole in the core with an absorber guide tube inserted in the lifting socket. …

A guide tube drive mechanism is provided. During refueling the drive mechanism lifts the guide tube 12 inches (six inches out of the core). …

The ultimate shutdown system is activated manually from a pair of ultimate shutdown buttons located in the remote shutdown facility (RSF), or from a similar pair of buttons in the RPS vaults. … (Reference [12])

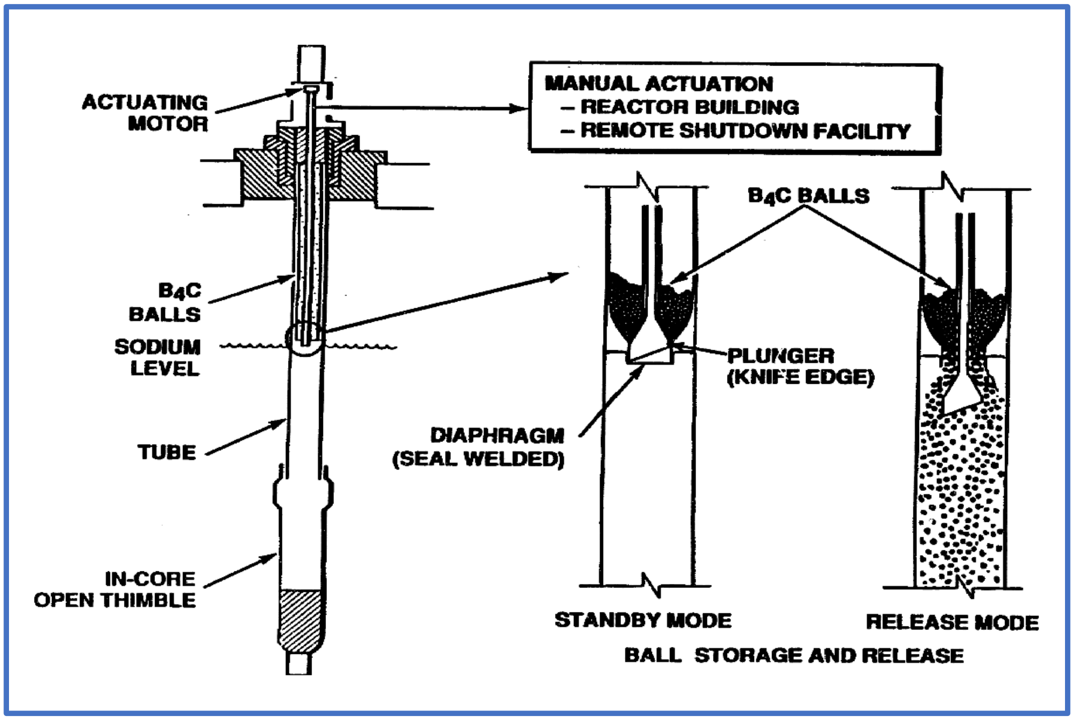


Figure C-1: Initial PRISM USS Assembly

[redacted]

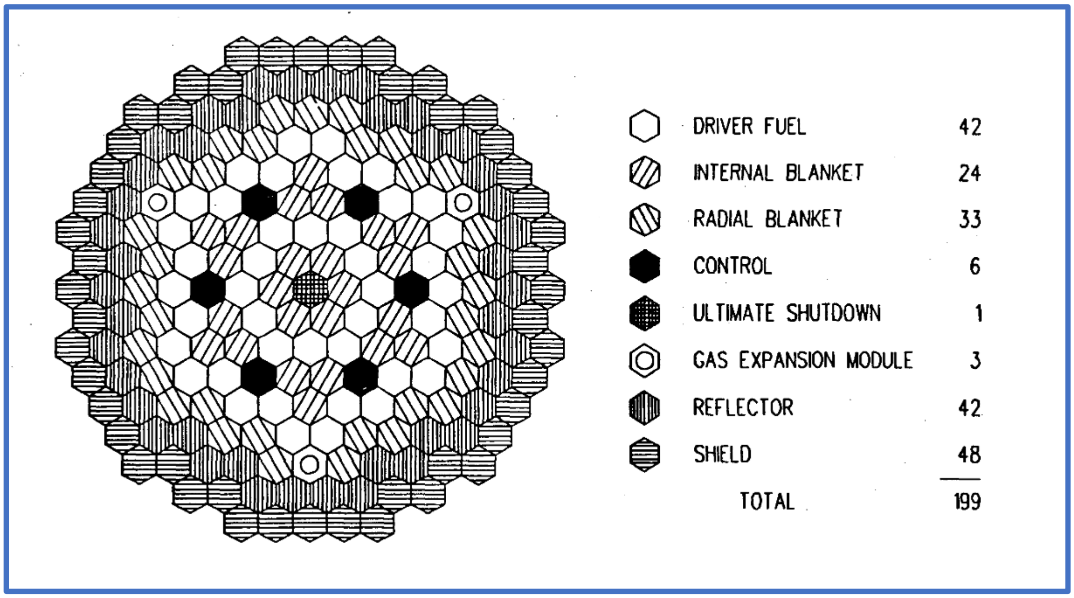


Figure C-2: Initial PRISM Core Configuration

The NRC performed a preapplication safety evaluation of the PRISM design and published their results in NUREG-1368 (Reference [13]). The results of the NRC evaluation of PRISM's initial USS are provided below.

4.6.5.4 Ultimate Shutdown System Activation

Concerns that the inherent core characteristics, while placing the core in a safe hot standby but still critical condition, would not take the core to cold shutdown led the pre applicant to incorporate an additional active shutdown system in the PRISM design. This USS is designed to release spheres of B_4C into a channel in the reactor core, which will bring it to a subcritical state, Similar to the liquid poison shutdown systems in LWRs, it is not as rapid as a control rod scram and is manually initiated, The inherent negative reactivity of the PRISM core would still play a role in an event in limiting the extent of the transient until the USS is activated. The worth of the USS absorber inserted into the core is sufficient to bring the reactor from 135 percent of full power to a cold shutdown.

The USS is activated from the RPS vaults or the shutdown facility (RSF). Unlike the control rods, the response of the USS must take into account delays associated with decisional protocol to activate the USS and in the transit time for an operator to proceed to the RPS vaults or the RSF to initiate the USS. Upper limits on the total to initiate must be determined in order to complete the transient analysis of the PRISM reactor con. It is possible that the staff may insist on a safety-grade actuation from the control room at a later date. (Reference [13])

A.2.2.2Ultimate Shutdown System (USS)

While the passive reactor shutdown mechanism, based on reactivity feedback, has significant safety advantages, it usually leaves the reactor in a critical condition and, therefore, exposed to further changes in system conditions. With the addition of the USS, GE has provided an alternate means of shutting down

[redacted]

[redacted] *reactor. USS activation causes many small spheres of B_4C to fall through a tube into the center of the core, in response to an operator-actuated shutdown command. This action results in a subcritical reactor producing only decay heat. The device fills an important gap in the PRISM safety defenses. That is, the passive shutdown no longer has to function indefinitely, because a neutronic shutdown can be anticipated within some reasonable time*. (Reference [13])

C.1.2 Revitalized USS

In 2006, GE Hitachi Nuclear Energy (GEH) revitalized the PRISM engineering effort (Reference [14]). The revitalized PRISM design differs significantly from the design that was evaluated by the NRC in the 1994 PSER. As shown by the discussion below, the reactivity control means, including the Ultimate Shutdown System (USS), were affected by the revitalized design.

> ***II.C. Reactivity Control***
>
> *Reactivity control for normal startup operation, load following, and shutdown is accomplished with control rods. Each control unit consists of a drive mechanism, a driveline, and a control assembly. A stepping motor, controlled by the plant control system, actuates a lead screw to insert and withdraw the B4C absorber. The nine control rods have scram diversity and shutdown redundancy. …*
>
> *The primary shutdown system is backed up by an ultimate shutdown system. These control rods use magnetic latches, which can be actuated by either the reactor protection system or automatically when the latch temperature exceeds the magnetic curie point temperature of the latch.*
>
> *Both shutdown systems are backed up by the inherent negative power reactivity feedback of the reactor core. This inherently negative reactivity feedback brings the core to a safe, stable power state following accidents*. (Reference [14])

Figure C-3 shows the revitalized PRISM core configuration, including the positioning of the three (3) USS control rods. Comparing Figure C-3 to Figure C- shows that the revitalized PRISM has substantially evolved from the initial PRISM design.

ARC looked for, but did not locate, documentation as to why GEH changed the USS from a B_4C ball based system to a B_4C rod based system. Nor were we able to locate any documentation regarding the NRC's evaluation of the revitalized PRISM USS.

[redacted]

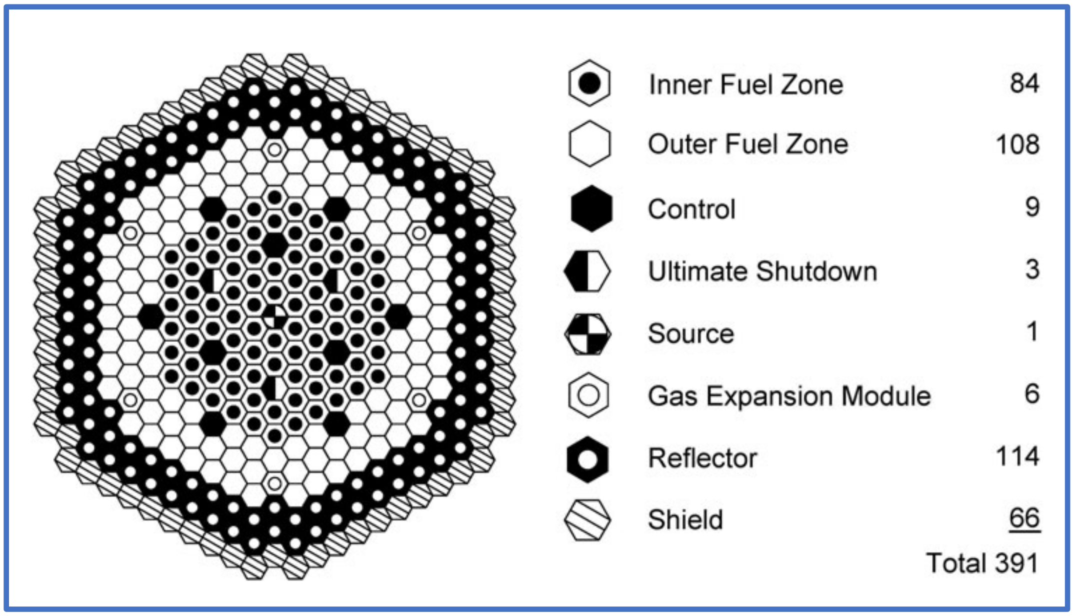


Figure C-3: Revitalized PRISM Core Configuration

C.2 Natrium Advanced Reactor — Independent and Diverse

The approach taken by TerraPower for the design of the two independent and diverse means of controlling the rate of reactivity changes for the Natrium advanced reactor is:

> *the Natrium design uses two control rod systems as independent and diverse means for reactivity control. The functions provided by the two means credited in Natrium PDC 26 [i.e., rod scram latch release versus rod motor drive-controlled insertion] are independent and diverse systems based on their design and function. Additionally, the geometric design of the two systems of control rods is different, providing protection against a common mode failure of rod binding. While this approach differs from the NRC staff rationale for adaptation in RG 1.232, it meets the intent of the GDC and ARDC requirements for reactivity control*. (Reference [10], Section 5.3.2)

TerraPower provided the proposed Natrium design to the NRC in the Reference [10] topical report (TR). The NRC provided the results of their review of the TR in Reference [11]. The entire NRC evaluation of the TerraPower proposed Natrium design for complying with DC 26 has been reproduced below.

> *4. Rationale for Meeting Natrium PDC 26*
>
> *TerraPower requested NRC staff review of the rationale provided in the TR for meeting the intent of Natrium PDC 26. In TR Section 5.3.2, "Independent and Diverse," TerraPower indicated that there are two different control rod assembly (CRA) designs as well as two different means of inserting the CRAs into the reactor core, namely a scram latch release that allows the CRAs to be pulled into the core by gravity and a separate motor driven scram insertion function that would allow the CRAs to be pushed into the core by the control rod drive system.*

[redacted]

Additional context on TerraPower's rationale for meeting the intent of Natrium PDC 26 was also provided during a November 10, 2022, meeting with the NRC staff (ML22301A073).

As discussed in SE Section 3.2.1, Natrium PDC 26 is based on SFR-DC 26 with a change to use SARRDLs instead of SAFDLs. The PDC requires a "minimum of two reactivity control systems or means," one of which must be "independent and diverse from the others." The NRC's rationale for adaptations to the GDC for SFR-DC 26 in RG 1.232 states that "the term 'independent and diverse' indicates no shared systems or components and a design which is different enough such that no common failure modes exist" between the systems relied on for sentence (1) and (3) and the system relied on for sentence (2) of the DC. The adaptations to the GDC also state that the system relied on for sentence (2) of the DC would be considered "important to safety but not necessarily safety-related."

TerraPower stated that it intends to demonstrate that, though it may not be consistent with the rationale in RG 1.232 discussed above, the combination of different CRA designs and means of insertion on a scram signal is sufficiently diverse and independent to meet the underlying intent and requirements for reactivity control. The argument relies in part on analyses performed using the Natrium probabilistic risk assessment (PRA) which indicated that the primary issues with the CRAs are related to binding, either associated with the gripper or binding between the bundle and the duct. The secondary control rod design intends to mitigate the bundle/duct binding failure mode that would otherwise be shared with the primary control rods, while the use of the motor driven scram insertion function would mitigate gripper binding. Various tests are planned to demonstrate the capabilities of the secondary control rod design.

The NRC staff generally finds this approach to be reasonable in meeting the underlying intent of PDC 26 because it uses a robust, RIPB approach to demonstrate that there is sufficient independence, diversity, and redundancy in the reactivity control system to reliably provide reactivity control and shutdown capability for normal operation, transients, and accident scenarios.

However, the NRC staff cannot reach a final determination on whether the proposed approach meets PDC 26 because it relies on integrated evaluations that involve the Natrium plant PRA, failure mode and effects analysis of the CRAs and control rod drive system, and the safety analysis. As such, though the approach is reasonable, the NRC staff is unable to make a final determination regarding whether TerraPower has demonstrated conformance of the Natrium design with PDC 26 at this time. The NRC staff will review conformance of the Natrium design with PDC 26 as part of a separate licensing action. (Reference [11])

C.3 Hermes Non-Power Reactor — Reactivity Control and Shutdown System

In December 2023 NRC issued a construction permit to Kairos Power LLC for construction of a non-power reactor, Hermes, in Oak Ridge, Tennessee. The Hermes reactor will be a 35-

[redacted] thermal (MWt) reactor using tri-structural isotropic (TRISO) fuel particles embedded in a carbon matrix pebble. The fuel particles will contain high assay low enriched uranium (HALEU). The reactor will be configured as a pebble bed with molten fluoride salt coolant. The reactor will use a functional containment implemented principally by the high temperature TRISO particle fuel.

The Hermes project developed project specific principal design criteria, including PDC 26 Reactivity control systems. The Hermes PDC 26 is identical to the RG 1.232, Appendix B, SFR DC 26, shown in Table 1-1, except that the term “design limits for the fission product barriers” has been changed to “specified acceptable system radionuclide release design limits” in criteria (1) and (2).

The description of the Hermes Reactivity Control and Shutdown System, taken from the Preliminary Safety Analysis Report is provided below.

> ***4.2.2 Reactivity Control and Shutdown System***
>
> *The reactivity control and shutdown system (RCSS) provide reactivity control during normal operation and also provides shutdown of the reactor in response to abnormal conditions or postulated events.*
>
> ***4.2.2.1 Description***
>
> *The RCSS inserts and withdraws control and shutdown elements to control reactivity in the reactor core. The system provides two separate means to accomplish this function - control elements and shutdown elements.*
>
> *The control elements are used to control the reactivity for normal operations and for planned, normal startup, shutdown, and power changes in the reactor. The control elements can be positioned throughout their range of travel to support operational demands. The portion of these elements that relate to their release and insertion on a reactor trip is safety-related.*
>
> *The shutdown elements, in combination with the control elements, are credited for shutting down the reactor during postulated events. The shutdown elements are located to optimize reactivity worth and to meet shutdown margin requirements. These elements have two positions, fully withdrawn or fully inserted. These elements are safety-related.*
>
> *Both the control and shutdown elements are tripped automatically by the reactor protection system, or manually from the main control room or remote shutdown panel. …*
>
> *The shutdown and control elements have two different designs. Each control element is an assembly of segmented annular cylinders. The annular cylinders are welded to connection plates at various points along the length of the control elements. Stainless steel spines are used to connect the array of control elements together. Each shutdown element is an array of small rods arranged in a cruciform shape.*

The Hermes control and shutdown element cross-sections are shown in Figure C-4.

[redacted]

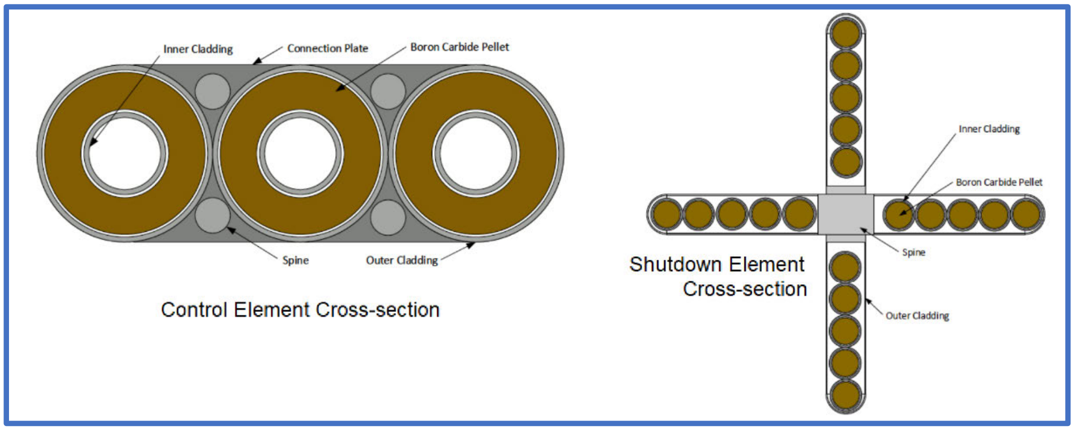


Figure C-4: Hermes Control and Shutdown Element Cross-sections

The NRC performed a safety evaluation of the Hermes design and published their results in Reference [20]). The results of the NRC evaluation of the Hermes Reactivity Control and Shutdown System are reproduced below in its entirety.

> *In the Hermes design, PDC 26 Conditions 1 and 3 address the safety related function of negative reactivity insertion used to evaluate the Chapter 13 postulated events (i.e., the SCRAM curve). In the Hermes design this safety related function is met by the insertion of the shutdown elements into the pebble bed reactor core. Condition 1 addresses the insertion of sufficient negative reactivity so the SARRDLs are not violated, and safe shutdown is achieved and maintained. Condition 3 is similar, but the SARRDL criterion is replaced with the capability to cool the core, which is less restrictive, followed by a safe shutdown after the postulated event (i.e., shutdown in long-term). For a power reactor application, Condition 3 was created to apply a less restrictive criterion to a less frequent event (i.e., accidents). For non-power reactors, the event frequency is not considered, and only postulated events are evaluated; hence, satisfying Condition 1 satisfies Condition 3. Kairos states in PSAR Section 4.5.3.1 that Condition 1 is met and hence so is Condition 3.*
>
> *The staff notes that PDC 26 Condition 2 states there should be a means, which is independent and diverse from others, capable of controlling the rate of reactivity changes from planned, normal power changes to assure the SARRDLs are not exceeded. Condition 2 refers to normal plant power changes using a method or system of reactivity control which differs from that used to satisfy Conditions 1 and 3. The staff notes that, in the Hermes design, the normal operation reactivity control function is satisfied by the control elements which are independent and diverse from the shutdown elements. The staff also notes that the control and shutdown elements are independent and diverse based on differences in (1) input signals (though both receive a trip signal from the RPS, which is conservative because control elements provide additional negative reactivity upon a trip), (2) the control mechanisms (motor-driven vs gravity), and (3) insertion geometries (reflector vs pebble bed).*
>
> *PDC 26 Condition 4 refers to keeping the reactor shutdown, typically at a lower temperature, such that plant maintenance, inspections, and refueling activities*

[redacted] *occur. The inserted negative reactivity should be sufficient to prevent a return to power at the conditions which correspond to these activities. In the Hermes design this is achieved by the safety related shutdown elements. To address PDC 26 Condition 4, PSAR Section 4.5.3.1 states that shutdown elements provide a means for maintaining the reactor shutdown to allow for interventions such as fuel loading, inspection, and repair.*

Based on its review, the staff finds that the preliminary level of detail provided on Hermes shutdown margin is adequate at this stage of the design and is consistent with PDC 26 because (1) the shutdown margin is sufficiently high, (2) the shutdown margin calculation methodology assumes the single most reactive control or shutdown element to be fully withdrawn from the core, which is conservative, (3) the element worth calculation methodology is sufficient, and (4) Kairos will perform source range control element worth testing. The staff also finds that the preliminary information is consistent with the relevant acceptance criteria from NUREG-1537, which states that the design should analyze and justify a minimum negative reactivity to ensure safe reactor shutdown. The staff notes that, while specific details of shutdown margin are not needed to issue a CP, the staff will review specific details of the shutdown margin in an OL application, so that the staff can review and verify the acceptability of the final design. (Reference [20])

[redacted]

Appendix D Evolution of Safety Rod Configuration

[redacted]

Safety Rod

Alternate Safety Rod

Figure D-1. Left: Initial safety rod design for ARC-100. Right: proposed alternate design.

Methodology:

The initial design of the safety rod, employed the identical configuration to that of the primary control rod For diversity of configuration that would significantly lessen the provability of common mode failure to insert the rods due to deformation affecting the external hexagonal duct of the control assemblies, it is desired to reduce the size of the pins in the safety rod assemblies and change the shape of its inner duct from hexagonal to another shape, in this instance a circular one.

The choice of configuration causes the the inner pin to be larger than the second ring of pins. Additionally, the second ring of pins has seven pins instead of the more conventional six pins. For ease of modeling, this was converted back to a six-pin design with equivalent B_4C and structural steel content (i.e., conserved material area from seven-pin design). The central pin was untouched, only the second ring modified. The assumed dimensions for this analysis are shown below. Notes for alternate model column below: “Pin Pitch” is for the equivalent six-pin design, ID/OD dimensions for the seven-pin design, except the equivalent six-pin design used a wire wrap OD of 0.1437 cm.

[redacted]

Table D-1. Dimensions used for both reference and alternate safety assembly designs.

	Reference Model		Alternate Model	
	Cold Dim. [cm]	Hot Dim. [cm]	Cold Dim. [cm]	Hot Dim. [cm]
[redacted]				

[1]7-pin alternate model spacing between pins in the second ring.

[2]7-pin alternate model distance from center of middle pin to center of second row pin.

[3]6-pin equivalent model pin pitch (for all pins).

[4]Equivalent circular diameter for 147.3 flat to flat inner hexagonal duct

The alternate model assumed the same pin cladding (0.07 cm cold) and inner duct thickness (0.3 cm cold) as the reference model. The same wire wrap thickness was used for all pins (except where noted above). Note that it is desirable to use a thicker wire wrap around the single central pin since it has a large gap between it and the other pins. This was not considered in the neutronics model, and it is not expected that having slightly more steel in the location would materially impact the boron enrichment estimate made later on.

Verifying the Geometry of the Alternate Design:

Spacing of pins in the inner duct needs to be taken into account. The distance to the second ring of pins from the inner ring (P_1) should account for the size of the inner pin diameter (D_i), an outer pin diameter (D_o), and the wire wrap diameter (D_w):

$$P_1 > 0.5D_i + 0.5D_o + D_w$$

$$P_1 > 0.5(5\ cm) + 0.5\ (4.4\ cm) + 0.133\ cm$$

$$P_1 > 4.833\ cm$$

The pitch between the second ring of pins (P_2) needs to accommodate the pins themselves and the wire wrap between the pins:

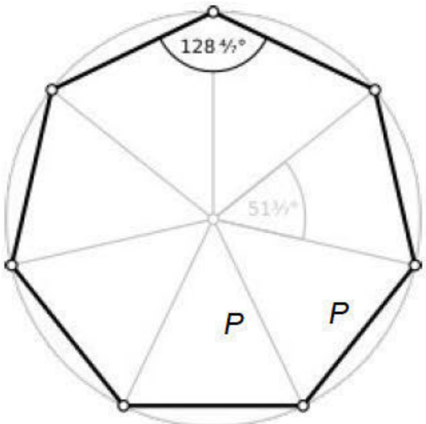
$$P_2 > D_o + D_w$$

$$P_2 > 4.4\ cm + 0.133\ cm$$

$$P_2 > 4.533\ cm$$

The more restrictive limitation is on P_2 for using a regular heptagon geometry:

[redacted]



<u>If tight on P_1:</u>

$$sec\left(\frac{128\frac{4}{7}}{2}\right) = \frac{P_1}{0.5P_2}$$

$$P_2 = 2P_1 cos\left(64\frac{2}{7}\right)$$

$$P_2 = 2(4.833\ cm)cos\left(64\frac{2}{7}\right)$$

$$P_2 \approx 4.194\ cm$$

Violates $P_2 > 4.533\ cm$

<u>If tight on P_2:</u>

$$sec\left(\frac{128\frac{4}{7}}{2}\right) = \frac{P_1}{0.5P_2}$$

$$P_1 = 0.5P_2 sec\left(64\frac{2}{7}\right)$$

$$P_1 = 0.5(4.533\ cm)sec\left(64\frac{2}{7}\right)$$

$$P_1 \approx 5.224\ cm$$

Satisfies $P_1 > 4.833\ cm$

We are in fact tight on P_2, so the outer pins will be close together with a gap between the inner pin. It is assumed that the pitch between the outer pins is 4.563 cm, as seen in the figure below. The corresponding pitch P_1 from the inner to outer pins is 5.258 cm. It is desirable to use a larger wire wrap diameter only for the inner pin to reduce the vibration of that pin.

The circular inner duct surrounding the eight control pins needs to be able to fit the second ring of pins. This means from the center of an outer pin, it needs to accommodate both the radius and wire wrap surround the pin. This inner duct inner radius (R_{DI}) is thus limited to:

$$R_{DI} > P_1 + 0.5D_o + D_w$$

$$R_{DI} > 5.258\ \text{cm} + 2.2\ \text{cm} + 0.133\ \text{cm}$$

$$R_{DI} > 7.591\ \text{cm}$$

$$2R_{DI} > 15.183\ \text{cm}$$

It is assumed that this inner sheath diameter is 15.199 cm, which satisfies the condition above. This is all summarized in the figure below.

[redacted]

[redacted]

Figure D-2. Dimensions of alternate design for ARC-100 safety assembly.

A notable takeaway is a significant reduction in the size of the inner duct. The reference model has a hexagonal outer flat-to-flat distance of 15.35 cm cold, whereas the alternate model has an outer diameter just below 15.80 cm cold. This dimension provides an average distance to the outer duct of 1.1cm (vs 0.91 cm) wit a maximum distance of 2.0 cm and a minimum distance of 0 675 cm.

Results:

The reactivity worth of the reference SR assemblies is -3469 pcm. It is expected that only making the geometric changes from the reference case to the alternate design and continuing to use natural boron (19.9a% B-10) will have less reactivity worth since the pin diameters were reduced, and this is indeed what is observed. To find the boron enrichment which is equivalent to the reference model, a few boron enrichments were selected and compared to the reference result. A boron enrichment of about **31.35 a%** B-10 would produce the same reactivity worth as the reference model using naturally-occurring boron (19.9 a% B-10). There are commercial sources of enriched boron – such as 3M corporation https://www.3m.com/3M/en_US/p/d/b40066327/.

Table D- 2. Reactivity impacts of different boron enrichments for safety assembly insertion.

Case	k-eff	ρ [pcm]
[redacted]		

[redacted]

Figure D-3. Reactivity impacts of different boron enrichments for safety assembly insertion.

APPENDIX E CHOICE OF B_4C BALLS SYSTEM

The choice of B_4C for the balls was made based on the very successful results obtained during operation of and the tests on both vented and unvented pins for their control rods. Almost without exception, the world's attention is on boron carbide control rods. Its relatively high absorption cross section, low cost, availability, ease of fabrication, and relatively low activation level makes it the ideal control rod material, and hence the material for the balls. Initially, the concern was whether control rods could be designed to achieve 50×10^{20} captures/cm^3. Rod sticking in foreign reactors also raised a concern related to overall functionality and performance. No degradation in SCRAM time or functionality has ever been observed with the FFTF control rod system. Figure D-1(taken from reference [6]) shows that this goal has been achieved and there have been no adverse consequences, and with over 2200 control rods irradiated with peak capture levels near 330×10^{20} captures/cm^3 (for 20% Enriched ^{10}Boron rods), and 220×10^{20} captures/cm^3 for operational rods

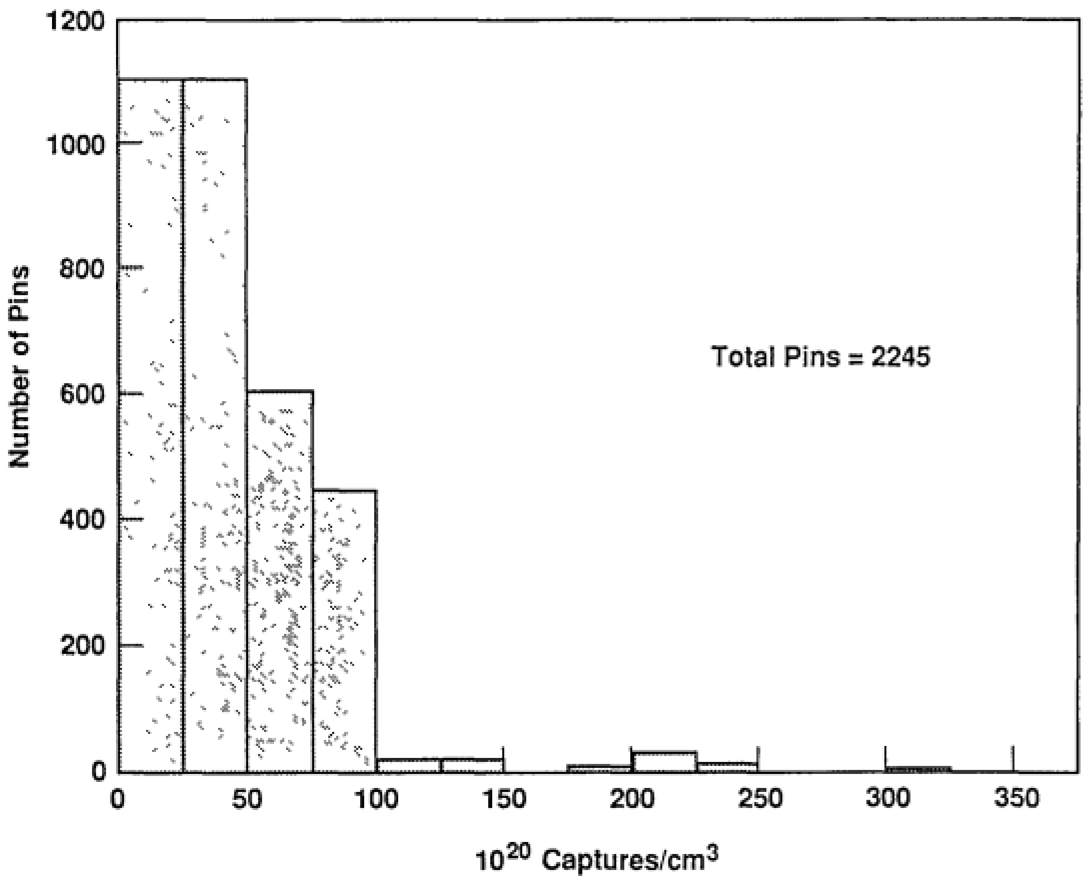


Figure D-1: Boron Carbide Absorber Pins in the Fast Flux Test Facility

For the carbide balls system, the irradiation period is expected to be short, until means are found to insert the control and safety rods. In addition after a relatively short time (which depends on the unprotected transient being experienced but would be at most several hours) the neutron flux is reduced by the inherent behavior of the reactor, which brings the core to stable controlled, but still critical decay heat levels after the primary and secondary pumps are

[redacted]

tripped therefore the balls that have fallen in the central assembly are subjected to a low flow rate of the sodium which is driven by natural circulation only, after the pumps cost down a two minutes is terminated by exhaustion of the batteries that provide them with power after the normal power is terminated.

Unlike the control and safety rods, which are continuously immersed in sodium flowing at relatively high velocity, the carbide balls will remain immersed for a substantially shorter period and at velocities that are quite low.

The carbide balls are nevertheless encased in a cladding of HT9, and helium released from the neutron irradiation (at decay heat levels) can cause a strain in the cladding. That strain can be reduced, but no eliminated by venting the cladding, allowing the helium to escape in the coolant.

From FFTF data the helium release quantity as a function of burn up and temperature indicated increasing fractional gas releases for burnups exceeding 100×10^{20} capture/cm^3 and temperature in excess of 1500°C [7], but notes that in general gas release fractions are 10% or less for conditions below those levels. Therefore, for the boron carbide balls, a release of 10% can be expected. This means that most of the helium is retained within the B_4C matrix, and this causes the B_4C to swell and a rate which is greater than the swelling of the cladding. A swelling correlation indicates a basic swelling rate of about 0.5% diameter increase per 10^{20} captures/cm^3.

To avoid significant strains of the cladding, an allowance of space is made , implicit on the assumption that the boron carbide balls have an equivalent “smear” density of 90%. Alternatively, if such provision is not made, failures of the cladding could occur and erosion of the B_4C by the flowing sodium could occur via the breach in the cladding to such erosion , and determined that even with large defects machined in the cladding, the B_4C loss was restricted to the pellet surface in the vicinity of the breach. In a loop test conducted in which a high burnup B_4C pellets were exposed to 538°C sodium flowing at 1.5 m/s, the total quantity of B_4C lost though a 2.5 mm wide slot defect was less than 4% of the local pellet volume after 50 days of test duration. The material that was washed out was very small in particle size (approximately grain-sized and would not be harmful to reactor components.

While such loss is small, it can be dtrimatal to the absorber function of the carbide balls. Irradiaiton causes deterioration in the physical integrity of B_4C, with the degree of degradation most severe at lower irradiation temperatures. Boron carbide pellets have frequently found to be cracked and fragmented after irradiation at high burnup levels, but this type fragmentation (with or without the presence of sodium) has not been observed to cause any degradation in absorber pin performance. Moreover pellets irradiated in sealed (helium atmophere) to levels of $80\text{X}10^{20}$ capture/cm^3 have been recover intact. Form the data on possible levels of washout in case of pin failure, and certain washout in case on no cladding, the choice has been made to provide HT9 cladding, with a thickness capable to withstand strains of the order of 2% or less.

The empty assembly hexagon inside its face (flat to flat) is 171.5 mm, and the inner diagonal distance is 198.03 mm. Assume that the maximum number of balls along the diaginal is 6, and eaach ball has a cladding of 0.5 mm thickness . The diamter of the ball with cladding would be limited to 39.5 mm, with the inner diameter of 38.5 mm. The volume of the inner ball is then $V = \pi/6\ d^3 = 29{,}880\ \text{mm}^3$. Epxerience at FFTF has shown that acceptable behavior of the B_4C material is obtained for 92% smear density. Therefore the diameter of the B_4C ball is 0.92 ×

[redacted] [redacted]

29,880 = 27,489 mm^3 ,resulting in a diamter of 37.44 mm of the carbide ball at full density. Therefore a ball of 39.5 mm OD has only an equivalent ball of full density B_4C material, of 37.44 mm. The rest is HT9 and helium

Therefore in the evaluation of the reactivity of the B_4C ball system, each ball is modeled as one of 38.5 mm in diameter filled with B_4C with 92% of theoretical density.

Figure D.2 shows that the B_4C volume (or packing) fraction for the 39.5 mm OD balls in the hexonagal duct is 60.4%.

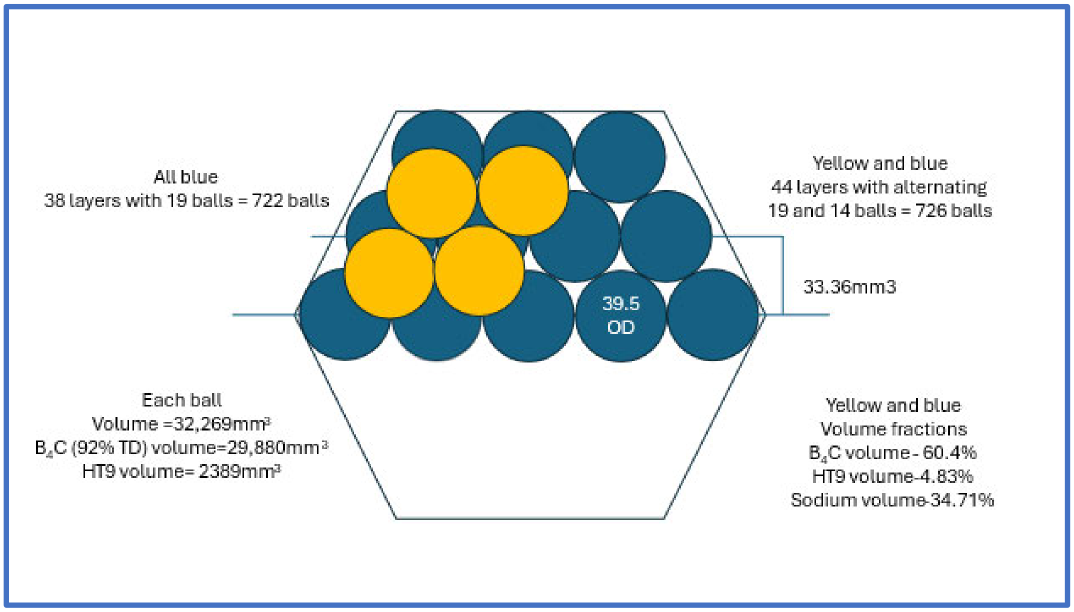


Figure D.2 Calculation of B_4C Volume (Packing) Fraction

[redacted]