Title:

ASSESSMENT OF THE PIUS PHYSICS AND THERMAL-HYDRAULIC
EXPERIMENTAL DATA BASES

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CONTENTS

List of Figures.....	i
List of Tables.....	i
Nomenclature.....	iii
Abstract.....	Section 1
Introduction.....	Section 1
Data Base Requirements and Needs - Certification Review Perspective	Section 2
Data Base Requirements and Needs - Code Adequacy Assessment Perspective	Section 3
Plant Description.....	Section 4
Representative Operational and Accident Sequences	Section 5
Key Systems, Components, Processes, and Phenomena.....	Section 6
Assessment of Existing and Planned Data Bases	Section 7
Assessment of Confirmatory Research Needs	Section 8
Code Requirements.....	Appendix A
Summary Tabulations	Appendix B

FIGURES

3-1	Code scaling, applicability and uncertainty evaluation methodology	Sec. 3, p. 8
4-1	PIUS PSID arrangement.....	Sec. 4, p. 8
5-1	NRC event category methodology	Sec. 5, p. 3

TABLES

3-1	Tabulation of PIUS Code Requirements.....	Sec. 3, p. 9
4-1	PIUS 600 Key Design Data	Sec. 4, p. 9
5-1	Representative PIUS Sequences	Sec. 5, p. 4
6-1	PIUS Systems, Components, Processes, and Phenomena	Sec. 6, p. 8
6-2	Key PIUS Systems, Components, Processes, and Phenomena	Sec. 6, p. 16
7-1	PIUS Separate Effect and Key Process Tests Completed.....	Sec. 7, p. 9
7-2	PIUS Separate Effect and Key Process Tests Planned.....	Sec. 7, p. 10
7-3	PIUS Integral Tests Conducted	Sec. 7, p. 12
7-4	PIUS Integral Tests Planned	Sec. 7, p. 14
B-3.2.1	PIUS Representative Sequences of Event Category I.....	
	(Anticipated Operational Occurrences)	App. B, p. 4
B-3.2.2	PIUS Representative Sequences of Event Category II	
	(Design Basis Events).....	App. B, p. 4
B-3.3.1.1	Data Types Useful for Confirming Key Phenomena in the Area	
	of Reactorfor Physics Representative PIUS Sequences in	
	Event Category-I.....	App. B, p. 5
B-3.3.1.2	Data Types Useful for Confirming Key Phenomena in the Area	
	of Thermal-Hydraulics for Representative PIUS Sequences in	
	Event Category-I.....	App. B, p. 6
B-3.3.2.1	Data Types Useful for Confirming Key Phenomena in the Area	
	of Reactor Physics for Representative PIUS Sequences in Event	
	Category-II	App. B, p. 8
B-3.3.2.2	Data Types Useful for Confirming Key Phenomena in the Area	
	of Thermal-Hydraulics for Representative PIUS Sequences in	

	Event Category-II.....	App. B, p. 9
B-3.4.1.1	Integral Data Types Useful for Confirming Key Phenomena in the Area of Physics for Representative PIUS Sequences in Event Category-I	App. B, p. 10
B-3.4.1.2	Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-I.....	App. B, p. 11
B-3.4.2.1	Integral Data Types Useful for Confirming Key Phenomena in the Area of Physics for Representative PIUS Sequences in Event Category-II	App. B, p. 14
B-3.4.2.2	Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-II.....	App. B, p. 15
B-3.5.1.1	Existence of Data Bases in the Area of PIUS Reactor Physics for EC-I Sequences.....	App. B, p. 19
B-3.5.1.2	Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-I Sequences.....	App. B, p. 20
B-3.5.2.1	Existence of Data Bases in the Area of PIUS Reactor Physics for EC-II Sequences.....	App. B, p. 24
B-3.5.2.2	Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-II Sequences.....	App. B, p. 25
B-3.6.1	Prioritized Areas of Missing or Partial Data for the PIUS Reactor in EC-I Sequences	App. B, p. 30
B-3.6.2	Prioritized Areas of Missing or Partial Data for the PIUS Reactor in EC-II Sequences	App. B, p. 32
B-3.7.1	Prioritized Areas for PIUS Modeling Enhancements for Phenomena of Importance in EC-I Sequences	App. B, p. 35
B-3.7.2	Prioritized Areas for PIUS Modeling Enhancements for Phenomena of Importance in EC-II Sequences	App. B, p. 36

NOMENCLATURE

ABB	Asea Brown Boveri
ALWR	Advanced light water reactor
AP600	Westinghouse-designed advanced light water reactor
AOO	Anticipated operational occurrence
BDBE	Beyond design basis events
BNL	Brookhaven National Laboratory
CFR	Code of Federal Regulations
CSAU	Code scaling, applicability, and uncertainty
DBE	Design basis event
ECC	Emergency core cooling
GE	General Electric
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MTC	Moderator temperature coefficient
NRC	United States Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PCRV	Prestressed concrete reactor vessel
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic risk assessment
PSID	Preliminary Safety Information Document
PWR	Pressurized water reactor
RES	Office of Nuclear Regulatory Research
SBWR	Simplified boiling water reactor

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Abstract

The PIUS reactor utilizes simplified, inherent, passive, or other innovative means to accomplish safety functions. Accordingly, the PIUS reactor is subject to the requirements of 10CFR52.47(b)(2)(i)(A). This regulation requires that the applicant adequately demonstrate the performance of each safety feature, interdependent effects among the safety features, and a sufficient data base on the safety features of the design to assess the analytical tools used for safety analysis. Los Alamos has assessed the quality and completeness of the existing and planned data bases used by Asea Brown Boveri to validate its safety analysis codes and other relevant data bases. Only a limited data base of separate effect and integral tests exist at present. This data base is not adequate to fulfill the requirements of 10CFR52.47(b)(2)(i)(A). Asea Brown Boveri has stated that it plans to conduct more separate effect and integral test programs. If appropriately designed and conducted, these test programs have the potential to satisfy most of the data base requirements of 10CFR52.47(b)(2)(i)(A) and remedy most of the deficiencies of the currently existing combined data base. However, the most important physical processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain rodged shutdown and control systems. For safety-related reactor shutdown, PIUS relies on negative reactivity insertions from the moderator temperature coefficient and from boron entering the core from the reactor pool. Asea Brown Boveri has neither developed a direct experimental data base for these important processes nor provided a rationale for indirect testing of these key PIUS processes. This is assessed as a significant shortcoming. In preparing the conclusions of this report, test documentation and results have been reviewed for only one integral test program, the small-scale integral tests conducted in the ATLE facility. Test design and data documentation were not requested from Asea Brown Boveri and, therefore, were not reviewed for the remainder of the separate effect and integral test programs that have been completed to date. Only brief outlines describing future programs have been provided and reviewed. The paucity of information about the future ABB testing programs is an important factor that must be considered when evaluating the conclusions presented in this document.

Section 1

Introduction

The advanced, passive reactor designs have a number of unique features that distinguish them from both the current generation of light water reactors (LWRs) and the evolutionary advanced light water reactors (ALWRs). All of the safety systems are designed to be passive, relying on natural forces, such as buoyancy and hydrostatic differentials, to drive safety injection flows. The stated objective of the passive plant design is to use passive systems to replace all the active, engineered safety systems presently used in LWRs and thereby enhance the reliability and safety of the next generation of nuclear power plants (Ref. 1-1). However, these designs also include nonsafety grade, active systems as the first line of defense in the event of transients.

The PIUS advanced passive reactor is a 640-MWe pressurized water reactor (PWR) based on a concept developed by Asea Brown Boveri (ABB). ABB submitted a Preliminary Safety Information Document (PSID) to the United States Nuclear Regulatory Commission (NRC) in 1990 for the purpose of obtaining a preapplication review of the design (Ref. 1-2). The PIUS reactor takes advantage of current PWR technology. Nevertheless, many features of this design are unique and will require a technical basis, which is different, in some respects, from the technical bases of other ALWRs currently under review. The Advanced Reactor Policy Statement (51 FR 24643) defines advanced reactors as those with innovative designs for which licensing requirements will be significantly different from the existing LWR requirements. The staff proposes to treat the PIUS design as an advanced reactor in accordance with the policy statement (Ref. 1-3).

An applicant seeking NRC certification of a design that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions" is required by 10CFR52.47(b)(2)(i)(A) to (1) demonstrate safety features through tests, analyses, or experience; (2) evaluate interdependent effects among safety features; and (3) establish adequacy of the data base for validating analytical tools. The NRC reviews the applicant's experimental and analytical programs, identifies concerns of potential safety significance for selected systems, and generates independent expertise and information for making timely regulatory judgments.

The NRC conducts research, as needed, to support its review of reactor designs submitted for design certification. Independent verification of the analytical and experimental information provided is necessary, especially in selected areas where the sole source of information is the applicant. The intent of research conducted by the NRC is to (1) ensure acceptability of vendor's codes, (2) provide an analytical audit capability, (3) identify deficiencies in the applicant's testing program, (4) identify additional testing needs, including confirmatory tests, and (5) identify major technical issues requiring early staff guidance (Ref. 1-4).

The principal products of thermal-hydraulic research are analytical tools (computer codes) used to understand and predict the plant response to deviations from normal operating conditions. The reliance on best-estimate codes to simulate the thermal-hydraulic behavior of a nuclear power plant is based upon three factors. First, it is too costly and unsafe to test the thermal-hydraulic response of severe events using operating or just-built nuclear power plants. Second, one cannot directly apply results from test facilities because of scaling compromises in their design. Third, the study of various plant recovery techniques can only be performed by computer codes (Ref. 1-5).

The ability of the computer codes to predict reactor response with an acceptable uncertainty is validated using: (1) basic experiments to derive empirical formulas for determining phenomena; (2) separate-effect experiments to test the code's predictions for a single, complex component; and (3) integral system tests that are used to evaluate the code predictions for a complete reactor. Experimental results are then compared with code predictions. The validated computer codes are used to evaluate design-basis accidents, the safety implications of actual events, and all hypothetical transient scenarios determined to be major contributors to risk as a result of probabilistic risk assessment (PRA) studies and actual operating events. The regulatory staff and NRC contractors use these verified computer codes to independently evaluate reactor designs and licensee submittals to ensure that designs and procedures enable operators to respond safely to a wide range of possible accidents and transients. This is done to ensure that analyses for licensing submittals performed by reactor vendors or utilities are accurate and that the analyses will result in designs and operating procedures that are capable of preventing severe accidents (Ref. 1-6).

Given the fundamental reliance on experiments that generate data needed for understanding phenomena, modeling plant components, and evaluating code capabilities for predicting the transient behavior of reactor systems, the NRC has determined that an early assessment of data base needs is appropriate for advanced LWR designs undergoing preapplication reviews. The NRC has contracted with Los Alamos National Laboratory to assess the experimental data bases for the PIUS reactor (Ref. 1-7). Specifically, Los Alamos is to (1) review data bases that have been used by ABB to validate its [coupled] steady-state and transient neutronic and thermal-hydraulic computer codes, (2) review other relevant data bases that may exist, (3) assess the combined data bases with regard to their quality and completeness for validating code features important to PIUS performance, (4) quantify the impact of deficiencies in the data bases upon important safety calculations, (5) identify areas where calculated results are so important that multiple, independent data bases should be used for validation, and (6) evaluate the need for confirmatory integral systems testing by the NRC for the PIUS design. The objective of this report is to document findings relative to tasks (1) through (6).

The remainder of this report is organized to logically present the information identified in the previous paragraph for tasks (1) through (6). Sections 2 and 3 consider data base requirements from two perspectives. In Section 2, data base requirements arising from the Code of Federal Regulations (CFR) and data base needs arising from NRC reviews of other ALWRs are identified. Guidelines used to establish the need for confirmatory testing are also summarized. In Section 3, data base requirements and needs that arise from providing data bases sufficient for assessing the adequacy of computer codes to be used in PIUS thermal-hydraulic and systems safety analyses are identified. Sections 2 and 3 identify important benchmarks against which the existing and planned PIUS experimental data base is evaluated. In Section 4, a description of the PIUS concept is provided and the key safety-related features of the design are discussed. Representative operational and accident sequences are described in Section 5. Key systems, components, processes, and phenomena in the PIUS reactor are discussed in Section 6. Ultimately, the need to accurately model and simulate the important transient and accident behavior (processes and phenomena) that will exist in the PIUS reactor (components and systems) during transients and accidents defines the features and capabilities that will be needed in computational tools used for PIUS safety analyses. These code features and capabilities, in turn, provide insights into the PIUS data bases that will be needed to satisfy the requirements of 10CFR52.47(b)(2)(i)(A). An assessment of the existing and planned data bases is provided in Section 7, considering the data base requirements identified in Sections 2, 3 and 6. Assessment of confirmatory research needs is provided in Section 8. Summary findings regarding the quality and completeness of the PIUS existing and planned data

bases to be used for validating code features important to PIUS performance are provided in Section 9. Observations concerning the gathering of additional data from existing experimental facilities and the need for new separate effects facilities are also provided. Areas where calculated results are so important that multiple, independent data bases should be used for validation are identified. Finally, the need for confirmatory integral systems testing by the NRC for the PIUS design is assessed.

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Section 2

Data Base Requirements and Needs - Certification Review Perspective

Data base requirements and needs arise from several sources. Section 3 identifies requirements that arise from the need to ensure that the safety analysis codes can simulate the key processes and phenomena in the systems and components that respond to the transients and accidents that determine the safety characteristics of the design. This section deals with regulations governing a plant design that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions." These requirements are found in the CFR.

There is an element of judgment applied by the NRC when determining the adequacy and completeness of data bases relative to the CFR requirements for a given reactor. Therefore it is instructive to review recent NRC precertification and certification activities as they apply to assessments of completeness and adequacy of experimental data bases for ALWRs. The expected outcome of such a review is the identification of qualitative standards, consistent with current review practices for ALWRs, that can be used to assess the adequacy of the PIUS reactor data bases. A related outcome is the identification of qualitative standards, consistent with current review practices for ALWRs, that can be used to assess the need for confirmatory integral systems testing by the NRC for the PIUS design.

Several advanced reactors are currently undergoing precertification or certification reviews per 10CFR52.47(b)(2)(i)(A), specifically, the Westinghouse-designed AP600 and the General Electric design of the simplified boiling water reactor (SBWR). The NRC has embarked upon a program of early reviews of significant new safety features with the goals of (1) identifying those new safety features that have a high likelihood of needing a lengthy verification program, (2) identifying missing information or potential weaknesses in the development and verification programs, and (3) recommending improvements in the applicant's testing or other verification programs, as needed, and initiating action on confirmatory NRC programs, if needed (Ref. 2-1).

Requirements from the Code of Federal Regulations

The requirements of 10CFR52.47(b)(2)(i)(A) mandate, in the absence of a prototype, the requirements for certification of a standard design that utilizes simplified, inherent, passive, or other innovative means to accomplish safety functions. The PIUS reactor is clearly subject to 10CFR52.47(b)(2)(i)(A). The applicant must adequately demonstrate that (1) "The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;" (2) "Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;" (3) "Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions;" and (4) "The scope of the design is complete except for site-specific elements such as the service water intake structure and the ultimate heat sink."

Insights From NRC AP600 Reviews

We first turn to the AP600 precertification and certification reviews for insights regarding the reviews of designs that utilize simplified, inherent, passive, or other innovative means to accomplish safety functions. These insights can be grouped into four categories: (1) design issues, (2) necessity for testing, (3) testing issues, and (4) important test facility features. Insights for each of these categories will be summarized in the following four paragraphs.

The AP600 design incorporates major changes in the design and operating philosophy for many plant safety functions. The functions of active safety-class systems in the current generation of plants are replaced by passive safety-class systems. The passive safety systems are different from the safety systems in the current generation of plants and do not have a proven performance history (Ref. 2-2). The staff is concerned about the performance of these new passive safety systems when they are incorporated into new primary loop geometries (Ref. 2-3). The passive systems rely upon small differences in temperature, pressure, or elevation for operation, and these may be affected by relatively minor variations in these parameters. All emergency core cooling (ECC) systems and the safety-related decay heat removal systems rely on gravity drain and natural circulation. These passive systems do not include pumps, and the system check valves operate on small pressure differentials. The staff also believes that the safety systems and nonsafety front-line systems can interact with one another. Reliability of both the active nonsafety systems and the passive safety systems will be crucial in finding the designs acceptable (Ref. 2-4). Because the operating philosophy of the AP600 calls for the nonsafety systems to serve as the first line of defense in transients or some accidents, nonsafety systems may be on line when safety systems are actuated, leading to untested systems interactions (Ref. 2-2).

Given the design issues and performance uncertainties described above, testing is recognized as the fundamental vehicle for addressing and reducing uncertainties to an acceptable level. Separate effect tests provide valuable data on individual components and systems (Ref. 2-2). Integral systems data are needed when legitimate technical issues arise that cannot be effectively resolved by component or separate effects testing, or by analysis (Ref. 2-3). In the absence of validated analyses that show conclusively that "safety system performance is adequate . . . concerns regarding systems behavior at high pressures should be resolved through performance of large-scale, full-height, full-pressure integral testing" (Ref. 2-2). Integral systems testing is needed to: (a) confirm new and unique features of the passive systems perform as claimed, and (b) uncover any adverse or unpredicted systems behavior or interactions (Ref. 2-2). Data are needed for the full range of conditions anticipated during the transients and events. For example, testing under both high- and low-pressure conditions was required for the AP600 to span the range of conditions over which the passive systems are expected to operate (Ref. 2-3).

In reviewing and commenting on the AP600 testing program, the NRC provided its perspectives on some important testing-program related issues. The staff stated, for example, that a systematic study of test facility scaling is important (Ref. 2-5, Enclosure 2). Further, the integral tests must study all important systems interactions, regardless of the safety classifications of the systems involved. Thus, the staff expresses concern when key safety systems are not included in the integral facility. The specific concern for the AP600 was that the passive containment cooling system (PCCS) was not coupled to a primary system simulation (Ref. 2-5, Enclosure 2). In addition to testing design basis event (DBE) conditions, some testing of beyond design basis events (BDBE) should be planned (Ref. 2-5, Enclosure 2).

A number of important test facility features were identified during the review of the AP600 testing program. Many of these would also apply to testing programs for other reactor types. Test facilities must be properly designed to minimize distortions in important phenomena and the behavior of key systems. Facilities should minimize scale distortions and consequent behavior that would be atypical of the actual reactors (Ref. 2-3). The staff provided the following insights regarding an AP600 integral test facility. An AP600 integral test facility would need to be large scale, full pressure, and full height. "Large scale" is a relative term, but in general means that the facility is large enough to minimize distortions in important phenomena and the behavior of key systems. The phenomena to be studied determine the type of scaling criteria that can be applied to determine the minimum size of the test loop that will not introduce unacceptable distortions that would make correspondence between test data and actual plant behavior difficult to determine. The staff stated that test facilities that are not full height are subject to additional distortions, particularly in natural-circulation systems (Ref. 2-5, Enclosure 2). Facilities should allow prototypic flow patterns to be established (Ref. 2-3).

Insights From NRC SBWR Reviews

The information describing NRC reviews of the SBWR testing programs was less extensive than for the AP600. Nevertheless, a number of useful insights were identified. As with the AP600, these insights can be grouped into four categories: (1) design issues, (2) necessity for testing, (3) testing issues, and (4) important test facility features. Insights for each of these categories will be summarized in the following four paragraphs.

The SBWR design incorporates major changes in the design and operating philosophy for many plant safety functions. The functions of active safety-class systems in the current generation of plants are replaced by passive safety-class systems. Even though some components are similar to current plants, operational conditions and the reliance placed on the components for the SBWR are considerably different. Thus, a number of issues must be resolved involving the performance characteristics of passive safety systems and components (Ref. 2-6, Enclosure 2). As with the AP600, a key issue is the potential for interactions among the various safety and nonsafety systems in the event of an accident (Ref. 2-6, Enclosure 2).

Given the design issues and performance uncertainties described above, testing is recognized as the fundamental vehicle for addressing and reducing uncertainties to an acceptable level. While recognizing that the SBWR safety system design is somewhat less complicated than that of the AP600 systems because of the lack of high-pressure injection systems, the staff noted that there is still the potential for interactions among the various safety and nonsafety systems. The staff stated that the preferred way to study interactive behavior is in integral tests, in which system components, configurations, and thermal-hydraulic conditions can be simulated with satisfactory accuracy (Ref. 2-6, Enclosure 2). The staff stated that all components critical to the operation of the passive safety systems should be tested (Ref. 2-6, Enclosure 2). This position was in response to a position by General Electric (GE) that it did not plan to test the valves, squib and check, in the passive safety system. The staff stated that at least one integral tests should include (a) all safety systems, (b) key nonsafety systems, and (c) relevant control systems (Ref. 2-6, Enclosure 2).

In reviewing and commenting on the SBWR testing program, the staff provided its perspectives on some important testing-program related issues. First, data must be

obtained in tests that come from facilities that are sufficiently large and appropriately scaled (Ref. 2-6, Enclosure 2). Second, the data must come from sufficiently prototypic testing. This issue arose several times in the review of the GE test program. GE presented the PANDA tests as "confirmatory" in nature rather than "developmental," and, therefore, not necessary to fulfill the requirements of 10CFR52.47 for analytical model verification and proof of system performance. However, the staff stated that previous testing was at much smaller scales and did not represent the current design of the passive heat removal or ECC system (Ref. 2-6, Enclosure 2). The issue of prototypicality was also addressed relative to the transport of boron into the core after activation of the Standby Liquid Control System (SLCS). GE asserted that data from the Advanced Boiling Water Reactor Program (ABWR) could be used to support the SBWR design. However, the staff stated that there were substantial differences in the two designs, including the ABWR's use of a pumped, rather than a passive, system (Ref. 2-6, Enclosure 2).

An important test facility feature related to one of the SBWR components was identified during the staff review of the testing program. The staff stated that the heat exchanger configuration in the facility was not representative of the full-size system, and that important effects, such as the distribution of noncondensibles in the heat exchanger header, could not be characterized because of the one-dimensional nature of the hardware tested. The staff concluded that a larger-scale simulation of the systems should be performed (Ref. 2-6, Enclosure 2).

Insights From NRC PIUS Reviews

The review record for the PIUS reactor is brief and reflects the early and preapplication status of the NRC's effort. Nevertheless, the staff has identified several issues related to this design. These are summarized in the following brief paragraphs based upon information provided in Ref. 2-7.

An evaluation of the expected performance and reliability of the density locks is needed. The operational performance of the reactor depends on the establishment and maintenance of a stable liquid interface in the density locks during normal operation and transients. The safety function depends on the certainty of rapidly breaking [activating] the density lock barrier. A solid basis for predicting qualitatively the behavior of these locks must be developed.

The experimental and analytic basis for analyzing the flow behavior around the gas baffle and other structures in the lower density lock needs to be evaluated. The structure over the lower density lock may cause significant variation in the flow velocity distribution in the core after the coolant pumps are tripped. This could adversely affect the delivery of a uniform boron density to all fuel assemblies simultaneously, possibly causing unacceptable transient and heat generation in the reactor.

The experimental and analytic basis for analyzing thermal convection and mixing in the pool needs to be evaluated. The pool around the reactor core provides cooling for the core, but uniform mixing is not certain because of the low exit velocities at the upper density lock. Flow at the upper density lock may not mix significantly with cool pool water before returning to the lower density lock.

The validity of preapplicant's analytical models, heat transfer correlations, and flow parameters for the core in the regime of low coolant velocity needs to be verified and validated.

Confirmatory Research Needs

As part of the data base assessment effort, the NRC requested that Los Alamos "evaluate the need for confirmatory integral systems testing by the NRC for the PIUS design." This is a difficult and challenging assignment because the applicant is required by 10CFR52.47(b)(2)(i)(A) to adequately demonstrate that (1) "The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;" (2) "Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;" and (3) "Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions."

Given these regulatory requirements on the applicant to perform "appropriate test programs" and ensure that "sufficient data exist on the safety features of the design," it is important that a rationale for additional or confirmatory testing by the NRC be identified. It is only within this framework that the need for confirmatory integral systems testing by the NRC for the PIUS design can be assessed. The needed rationale has, indeed, been developed and articulated by the NRC. Key elements of the rationale have most recently been articulated during the precertification and certification reviews of the AP600 and SBWR, but many of the ideas date from earlier reviews.

In the following paragraphs, the reasons for which the NRC performs confirmatory research are identified, and the appropriate citations for this information are provided.

1. In striving to develop the information required to support design certification, the applicant performs tests and analyses. The NRC has concluded that *independent* tests and analyses are needed to ensure that the safety-related characteristics of the design are identified and understood. Thus, the concept of independence plays an important role in the rationale for confirmatory testing. Key supporting ideas are as follows. Confirmatory testing allows the staff to develop an independent data base for assessing the safety performance of the advanced passive designs (Ref. 2-3, Enclosure 1). Confirmatory testing permits an independent assessment of the capability of proposed cooling systems (Ref. 2-8). Confirmatory testing supports an independent assessment of the designs (Ref. 2-8). Confirmatory testing provides additional information that is needed to independently confirm a vendor's analysis, including key assumptions used in safety evaluations (Ref. 2-8). The NRC lacks experimental data on the performance of the new passive safety systems incorporated into new primary loop geometries in an integral loop configuration (Ref. 2-3).
2. In the absence of confirmatory data, the only data base available for new reactor types is that generated by the applicant. The NRC believes that it is important to have a *diverse* data base (Ref. 2-8). Implicit to applications of best-estimate codes to analyses of postulated transients and/or of accident events in nuclear power plants, is the assumption that these codes have the capability to scale up phenomena and processes from test facilities to full scale plant conditions. To demonstrate this crucial capability, best-estimate codes were assessed against experimental data from facilities of different dimensions. As a result, the NRC planned, designed and conducted both separate effects and integral effects tests in facilities of different scales (Ref. 2-9). Thus, confirmatory testing broadens the data base in a significant way; it provides data from one or more additional facilities at different scales. In addition, it is possible that the confirmatory research facilities might be designed to a different scaling rationale (Ref. 2-3, Enclosure 1). The staff has concluded that integral testing

in a facility of different scale and scaling rationale provides a diverse data base and is prudent for independent confirmation of code predictions of passive safety system performance (Ref. 2-8).

3. The NRC has developed its own computer codes which are used for independently assessing and auditing the plant performance. The NRC's Office of Nuclear Regulatory Research (NRR) has stated its need for an analytical audit capability (Ref. 2-10). The NRC states that experience has shown that the staff should perform its own integral system testing in order to provide data against which these codes could be independently validated (Ref. 2-3). The extent of the validation of the NRC's analytical codes for the advanced designs is considered substantially less than it is for currently operating reactors. The NRC has a limited data base for validating current thermal-hydraulic models over the range of operating conditions that could be experienced in these plants (Ref. 2-3). Confirmatory testing allows the staff to investigate performance issues related to code validation, which may not have been addressed by the applicant (Ref. 2-3, Enclosure 1).
4. The applicant's safety case is reviewed and evaluated by the NRC staff. The staff is responsible for determining and certifying the adequacy and acceptability of the safety case relative to the requirements of 10CFR50. To perform the required review and certification, the staff must be knowledgeable about the design and its safety characteristics. It is difficult to develop the requisite level of understanding by only reviewing the documentation submitted by the applicant. The NRR has identified support needed from the NRC's Office of Nuclear Regulatory Research (RES) for NRR's design certification review. Support was requested to (1) ensure acceptability of the applicant's codes, (2) identify deficiencies in the applicant's testing program, (3) identify additional testing needs, including confirmatory tests, and (4) identify major technical issues requiring early staff guidance (Ref. 2-10).

A key element in the rationale for confirmatory research is providing the opportunity for the staff to develop the experience base so that safety reviews and assessments of the requisite quality can be performed. Key supporting ideas are as follows: (1) Experience obtained in developing a confirmatory testing capability will be of significant benefit in aiding the staff's assessment of the vendor testing programs (Ref. 2-3, Enclosure 1). (2) Confirmatory testing leads to an improved understanding of atypicalities in the applicant's test facilities (Ref. 2-8). (3) Confirmatory testing allows the staff to gain insights into the design and operation of large-scale thermal-hydraulic test programs directed to study passive system performance (Ref. 2-3, Enclosure 1). (4) Confirmatory testing allows the staff to develop an independent experience base for assessing the safety performance of advanced passive designs (Ref. 2-3, Enclosure 1). (5) The staff will develop an improved ability to conduct audit analyses and predict the safety behavior of new designs (Ref. 2-3, Enclosure 2). (6) Confirmatory testing provides deeper understanding of integral plant behavior and enhances staff understanding (Ref. 2-10).

5. There is an element of uncertainty with new designs, particularly with designs that depart in significant ways from the experience base accumulated through the years. All designs contain margins to ensure that safety limits are not exceeded. However, it is highly desirable that these margins be assessed. Confirmatory research supports the verification of margins used by the staff in its regulatory positions and acceptance criteria for the design (Ref. 2-3). Confirmatory testing will allow the staff to reduce uncertainties that may exist in its current thermal-hydraulic codes, and improve the ability to conduct audit analyses in predicting the safety behavior of the new designs

(Ref. 2-3, Enclosure 1). A confirmatory testing capability also provides the capability to conduct follow-up investigations on performance issues that arise later in the design review process (Ref. 2-3, Enclosure 1).

6. In addition to supporting the identification of uncertainties and margins for known phenomena, confirmatory testing can also address unpredicted systems interactions. The staff have concluded that questions will frequently remain regarding the potential for unpredicted systems interactions, even after the applicant has completed its test program (Ref. 2-3). Confirmatory testing provides the NRC with the capability for exploring a range of conditions and resolving questions about unpredicted systems interactions. The NRC performs exploratory [confirmatory] research to identify potential safety concerns related to advanced reactor designs (Ref. 2-8).
7. It is the responsibility of the applicant to provide appropriate systems data to demonstrate acceptable safety systems performance for transients up to the plant design basis. Therefore, an NRC-sponsored confirmatory program may not need to simulate the plant in detail. Rather, key safety systems and aspects of the systems configuration should be represented. In such a facility, parametric studies of systems behavior and phenomenological issues over a range of conditions up to and *beyond the design basis* can be tested (Ref. 2-3, Enclosure 1).
8. The NRC also notes that public confidence is enhanced when confirmatory testing is conducted by an organization that is independent of the applicant. The NRC provides its view in a historical framework and states that "in addition to the very substantial improvements that were made in the analysis codes that predict reactor accidents, the operation of these loops produced the less tangible but very important benefit of improved public confidence in the NRC staff and nuclear industry regarding our understanding of reactor accident behavior" (Ref. 2-3, Enclosure 2).

It is appropriate to conclude on a cautionary note. The elements of the rationale for confirmatory testing tabulated in the previous paragraphs seem convincing. However, the specific justification for each confirmatory program must be well defined and documented. This point was clearly illustrated when the Advisory Committee on Reactor Safeguards (ACRS) commented on the NRC's confirmatory research plan for the AP600. The ACRS stated that the staff should further define not only its concerns, but also identify what information must be obtained in order to allay those concerns and allow licensing action to proceed (Ref. 2-3, Enclosure 4).

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Section 3

Data Base Requirements and Needs - Code Adequacy Assessment Perspective

In Section 2, we identified data base requirements arising from the CFR, and data base needs arising from NRC reviews of other ALWRs. Data base requirements and needs can also be identified from a second and equally important perspective—the necessity of having safety analysis tools (codes) that adequately simulate the steady-state, transient, and accident behavior of each design undergoing certification review. Thus, data base requirements arise from the need to ensure that the safety analysis codes can simulate the key phenomena and processes in the components and systems that respond to the transients and accidents that determine the safety characteristics of the design.

The importance of the linkage between the safety analysis codes and the data bases needed to assure the adequacy of those codes cannot be over emphasized. An applicant's demonstration of the safety of a nuclear power plant includes the analysis of plant response to postulated disturbances in system performance parameters, and to postulated malfunctions or failures of equipment. These safety analyses are one of the main focuses of the Commission's licensing review (Ref. 3-1). The regulatory staff and NRC contractors use verified computer codes to independently evaluate reactor designs and licensee submittals to ensure that designs and procedures enable operators to respond to a wide range of possible accidents and transients. This is done to ensure that analyses for licensing submittals performed by reactor vendors or utilities are accurate and that the analyses will result in designs and operating procedures that are capable of preventing severe accidents (Ref. 3-2).

Identifying data base requirements by evaluating the model development and validation requirements of safety analysis codes is a challenging endeavor. A large number of phenomena and processes, some of which are complex, are expected to occur in the PIUS reactor. If each code model for each system, component, process, and phenomena is of equal importance in determining the transient and accident behavior of the PIUS reactor, the effort and cost to develop the needed data base might be prohibitive. However, experience with the current LWRs has shown that this is not the case (Refs. 3-3, 3-4). Rather, the processes and phenomena occurring in some of the components and systems are more important, and they determine the course and outcome of a transient or accident sequence. Thus, it is prudent to identify those systems, components, processes, and phenomena that have the greatest impact on the outcome of transient and accident scenarios in advanced, passive reactor designs. This information can then be related to the models and correlations in the code, which in turn can be used to prioritize data base requirements. For future reactors, experience and expert judgment must be employed because a sizable testing and operational data base does not exist. A structured and scrutable approach for conducting critical evaluations was developed by the NRC and its contractors and was demonstrated for a large-break loss-of-coolant accident (LOCA) in a Westinghouse four-loop PWR. This methodology is named the code scaling, applicability, and uncertainty (CSAU) evaluation methodology (Ref. 3-3). A brief discussion of this methodology follows to highlight those parts of the CSAU process that can be used to develop insights into PIUS data base requirements and needs from the code adequacy assessment perspective.

CSAU Overview

The CASU evaluation methodology consists of three primary elements as shown in Fig. 3-1. For a future reactor, the steps in the first element have the greatest applicability in identifying data base requirements and needs from the code adequacy perspective. Only the activities in this element will be summarized.

As described in Ref. 3-3, the first element focuses on "Requirements and Code Capabilities." A particular reactor design, and the scenarios to be examined in that design, are selected. The processes and phenomena occurring in the plant components and systems are then listed, the processes and phenomena are ranked, and a phenomena identification and ranking table (PIRT) is developed. The need for such a ranking process arises from the fact that the resources required to develop code capabilities and supporting data bases for every process and phenomenon occurring during the selected transient are too large. Furthermore, although many processes and phenomena occur during a given transient, the overall plant behavior is not influenced equally by each. Consequently, a sufficient and more efficient (cost-effective) approach is to rank processes and phenomena by evaluating their importance relative to a set of primary safety criteria so that only the significant contributors need to be evaluated. For current reactors with an available experimental and operational data base, a list of the processes and phenomena occurring during the selected transient is compiled following examination of these data bases. For planned reactors, calculations of key transients can be examined to identify the processes and phenomena. Of course, the uncertainty is increased as less well-developed data bases are used.

Given the central focus of safety analysis codes in safety assessment, the next efforts concentrate on the selected analysis code. Code applicability is assessed by reviewing the adequacy of the individual models and correlations in the code and by reviewing the adequacy of the integrated code. These two adequacy assessment reviews are summarized in the following section. These reviews are conducted with reference to the key phenomena and processes identified in the PIRT. There are several outcomes of the adequacy reviews. First, missing or deficient code models are identified. If a separate effects data base currently exists to support development of the missing model or improvement of the deficient model, development can proceed. However, if the data base is missing or inadequate, then requirements for separate effect testing are generated. Second, if an integral effects data base exists, assessment of the integrated code can proceed. However, if the integral effects data base is missing or inadequate, requirements for integral testing are generated. In either case, only those separate effect and integral effect tests necessary to support code needs for the most important processes and phenomena identified in the PIRT are required.

Code Adequacy Assessment

For each analytical model, either stand-alone or embedded in a computer program, the following information should be reviewed and evaluated to determine the technical adequacy of the model: (1) pedigree, (2) applicability, and (3) fidelity (Ref. 3-5).

An integrated method will frequently contain a number of individual analytical or correlation-based models. The pedigree of each model is examined by considering the completeness and adequacy of information discussing (1) model origin including appropriate citations; (2) model basis and physical reasonableness; (3) assumptions, simplifications, restrictions, and limitations; (4) original application; and (5) experimental or other supporting data bases.

The applicability of each model is examined by considering the completeness and adequacy of information discussing (1) key or dominant phenomena from the PIRT, which are modeled; (2) identification of the model used for each dominant phenomena; and (3) justification for usage of the model beyond the original pedigree, if appropriate, including the use of the model beyond its initial scale of origin.

The fidelity of each model is determined by examining the prediction of key phenomena and assessing if the prediction is acceptably good. A key process for certifying the adequacy of a method is to consider the completeness and adequacy of the assertion that the method has been validated against an appropriate set(s) of separate effect test experimental data. Acceptable fidelity can be demonstrated if the model correctly predicts the major trends of the phenomena being modeled, the uncertainty can be quantified, and the uncertainty is acceptably small. A secondary method of certifying the adequacy of a model is to compare the completeness and adequacy of documented comparisons with other models previously found to be adequate.

In a system analysis method, individual models are usually integrated into a computer program or code. Applicability and fidelity should be reviewed and evaluated for each integrated method or code. The following descriptions are provided to describe important areas of review focus.

The applicability of an integrated analytical method rests on four elements (Ref. 3-5):

- (a) The field or conservation equations that provide the capability to address global processes. The basis for the governing equations should be reviewed and appropriate citations examined. Assumptions, simplifications, restrictions and limitations should also be reviewed.
- (b) The closure equations that provide the capability to model and scale particular processes. By reviewing the discussion for each individual analytical model as discussed above, item (b) will be satisfied.
- (c) The numerics used in the integrated analytical method. The capability of the numeric scheme used to perform reliable calculations is examined by considering the completeness and adequacy of information provided. Particular attention should be given to features that may restrict code applicability such as numerical diffusion and lack of mass or energy conservation.
- (d) Structure and nodalization that address the capability of the method to model plant geometry and to perform accurate integral calculations. User guidelines for noding and modeling that, if followed, ensure applicability of the integrated method are examined by considering the completeness and adequacy of information provided.

Fidelity of the integrated analytical method or code is based upon the demonstration that the representation or prediction of the collection of key phenomena in an accident sequence or event is acceptably good. Acceptable fidelity can be demonstrated if: (1) the integrated method correctly predicts the major trends of the dominant phenomena throughout the transient(s) being considered, (2) the uncertainty can be quantified, and (3) the uncertainty is acceptably small. A key process for certifying the adequacy of a method is to consider the completeness and adequacy of the assertion that the method has been validated against an appropriate set(s) of integral test experimental data. A secondary process for certifying

the adequacy of a method is to compare the completeness and adequacy of documented comparisons with other methods.

PIUS Data Base Requirement and Needs

In previous parts of Section 3, a structured approach for identifying PIUS data base requirements and needs from a code adequacy perspective have been described. Determination of data base requirements using this approach does require a significant effort by experienced and qualified personnel. At present, however, the PIUS design concept has only been submitted for preapplication review and allocation of the needed resources for the structured, and detailed review described previously is not warranted. Rather, extensive code validation and documentation are not required at this stage because the review is preliminary. Also, time does not permit such validation and documentation (Ref. 3-6).

Given this circumstance, a less time-intensive approach has been taken to identify (1) code requirements and (2) related data base needs. Both ABB and the NRC have noted that the PIUS reactor takes advantage of current PWR technology. The NRC also noted that many features of the PIUS design are unique and would require a technical basis that is different than those previously considered in the licensing reviews (Ref. 3-7). This logic is adopted for the remainder of this section by first summarizing historical code requirements for current LWRs and then addressing the unique PIUS components and phenomena that lead to other code requirements.

PIUS Safety Analysis Code Requirements

Los Alamos has recently completed a detailed assessment of code requirements for New Production Heavy Water Reactor (NP-HWR) system neutronics and thermal-hydraulics (Ref. 3-5). Although there are several unique PIUS components and phenomena, we believe that it is possible to provide an initial tabulation of code requirements based upon our experience in developing the NP-HWR code requirements.

This tabulation is provided as Table 3-1. Each item in Table 3-1 is further summarized in Appendix A. The detailed justification and appropriate references for each requirement identified in Table 3-1 and summarized in Appendix A are provided in Ref. 3-5. Although the detailed justification statements are based, in part, upon NP-HWR requirements, many of the justification statements were based upon NRC involvement in licensing activities for the commercial light water reactors. Therefore, we believe that Table 3-1 represents an appropriate initial tabulation of requirements that must be satisfied by codes used to predict the thermal-hydraulic and systems behavior of the PIUS reactor during normal operation, anticipated operational occurrences (AOOs), DBEs, and the initial phases of BDBEs while the core is still intact.

PIUS Components and Systems

The PIUS reactor concept includes a number of components that are based upon well established LWR technology. These include:

- Standard PWR fuel, but of shorter length
- Once-through steam generators
- Primary coolant pumps
- Relief and safety valves
- Hot- and cold-leg piping
- Containment structure of pressure-suppression type

- Drywell
- Wetwell
- Concrete pressure vessel
- Turbine plant
 - Turbine
 - Generator
 - Steam bypass system
 - Turbine condenser
 - Condensate and feedwater
 - Circulating water system

While noting the similarity of these components to those in existing LWRs, a caution is appropriate. With reference to the SBWR, the NRC has stated that, "Despite the similarity of some SBWR components, such as safety/relief valves and the isolation condenser, to those in current plants, operational conditions and the reliance placed on the components for the SBWR are considerably different from current generation plants. As a result, there are a number of issues that must be resolved involving the performance characteristics of passive safety systems and components" (Ref. 3-8).

Accurate modeling of the primary coolant circuit, which includes the core, hot- and cold-legs, steam generators, and primary coolant pumps, will be necessary. The data bases for these systems and components must be reviewed to determine if they are adequate given the unique PIUS configuration, especially where they are incorporated in the passive safety systems that characterize this design. The linkage between the pressurizer component, pressurizer relief valve system, and pressure-suppression dry and hot wells is one such system that deserves close examination.

The PIUS reactor concept also includes a number of components that are unique to this reactor. These include:

- Lower density lock
- Upper density lock
- Scram-valve system
- Reactor module riser and downcomer arrangement
- Reactor pool
- Siphon breaker
- Pressurizer arrangement

One component is notable by its absence: the PIUS reactor has no control rods for power control. Reactivity control is by boron concentration and coolant temperature via the negative moderator temperature reactivity coefficient.

The intrinsic and passive safety features of the PIUS design have been selected with the objective of making the possibility of a core degradation accident extremely small.

The PIUS reactor does not have control rods. However, two independent liquid boron reactivity control systems are provided. The normal reactivity control system pumps boron into the primary coolant loop to control reactor power or effect a reactor shutdown; this system is safety grade only within the bounds of the containment isolation valve. A scram-valve system consisting of four loops, one for each coolant loop, is connected from the reactor pool inside the steel vessel extension to the steam generator outlet plenum chamber on the suction side of the reactor coolant pumps. However, this system depends upon the continuing operation of the nonsafety-grade primary coolant pumps. A fully safety grade, intrinsic, and passive scram function is also provided for PIUS. This reactivity control

system relies on the ingress of highly borated water through the lower or upper density lock from the reactor pool to scram the reactor. This ingress occurs when the equilibrium conditions across the thermal barrier of the density locks are disturbed by an imbalance between the thermal core heat generation and removal rates.

Accurate modeling of the PIUS reactivity control systems requires the capability to simulate (1) the physical characteristics of the scram valves and associated piping, (2) the physical characteristics of the lower and upper density locks, (3) the control of the thermal barrier across the density locks, (4) the dynamic performance of the density locks when equilibrium conditions are upset, (5) the transport of boron from the reactor pool into the primary coolant system whether through the scram-valve system or the density locks, (6) the core neutronic response to the introduction of boron into the core including multi-dimensional effects and fluid-neutronic interactions such as oscillations. An experimental data base will be required to support modeling of the reactivity control systems, components, processes, and phenomena.

The PIUS reactor normally uses nonsafety-grade, forced cooling systems for the primary coolant and the reactor pool. Should these systems become unavailable, fully safety-grade, intrinsically passive cooling systems are also provided. Each of these systems relies on natural circulation. The core is cooled as a natural-circulation cooling circuit between the reactor pool and the primary circuit via the continuously open lower and upper density locks. In the event that the path through the lower density lock is blocked, a natural-circulation cooling circuit located between the reactor pool and the primary circuit via the upper density lock and the pool stand pipes at the top of the reactor module will operate. Should the upper density lock be blocked, the natural-circulation path between the reactor pool and primary circuit would be via the lower density locks, and the pool stand pipes at the top of the reactor module will operate. The reactor pool is not the ultimate heat sink for core-generated heat. A fully safety grade, intrinsically passive pool cooling system is provided. A number of coolers are immersed in the pool and are located along its perimeter at the level of the upper density lock. These coolers are connected to natural-draft cooling towers located at the top of the reactor building.

Accurate modeling of the PIUS inherently passive, natural-circulation systems is necessary. Safety analyses codes used to model PIUS transient and accident behavior must have the capability to model the linked components that constitute the natural-circulation flow paths. Pressure losses must be accurately simulated. Multidimensional effects, if important, must also be simulated. The natural-circulation flow rates must be accurately calculated, not only for determining that the core is adequately cooled, but also to determine the amount of boron that enters the core from the reactor pool. Because multiple flow paths exist, the relative magnitudes of flows through these paths must be accurately simulated as well as the mixing of the fluid streams where they join. An experimental data base will be required to support modeling of the natural-circulation systems, components, processes, and phenomena.

A siphon breaker is provided in the design to ensure that the primary system and reactor pool inventories are not lost through a siphoning action during a LOCA. If the siphon-breaking function is an important process, accurate modeling of the siphon breaker will be necessary to ensure the siphoning effect is accurately modeled. At the present time, BB believes that the siphon may be broken naturally, whether a siphon-breaker component is provided or not. An experimental data base will be required to support modeling of the siphon-breaker performance, if it is determined to be an important component that affects the path and outcome of key processes.

The pressurizer configuration in PIUS differs from the current LWRs. This component provides pressure control for both the primary system and the reactor pool. It provides a potential flow path between the reactor pool and the primary system. The pressurizer system also contains a spray system. The pressurizer control system will function during normal operation and some transients. It will provide a potential circuit for fluid flow during all transient and accident conditions. Accurate modeling of the key safety-related pressurizer processes and phenomena is necessary.

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Fig. 3-1.
Code scaling, applicability, and uncertainty (CSAU) evaluation methodology.

Table 3-1

Tabulation of PIUS Code Requirements

General Design Features

- Best-Estimate Methodology
- Thermal-Hydraulic Capabilities
- Adequate Numeric Performance
- Total Analysis Costs

Modeling Capabilities

Component Modeling

- Fuel Assemblies
- Core
- Plena
- Reactor Coolant Pumps
- Steam Generator/Heat Exchangers
- Vessel including Riser and Downcomer
- Pressurizer
- Pipes
- Valves
- Heat Structures
- Upper Density Lock
- Lower Density Lock
- Siphon Breaker

Systems Modeling

- Component Integration
- Control
- Plant Protection

State Modeling

- Steady
- Transient

Multidimensional Modeling

Reactor Kinetics Modeling

Fluids and Materials Property Data

- Light Water, Single and Two Phase
- Noncondensable Gas
- Structural Materials

Key Phenomena Modeling

Functional and User Interface

- Documentation
- Adaptability and Maintainability
- Ease of Use and Interpretation

Verification and Validation

Quality Assurance

Section 4 Plant Description

Overview

The PIUS advanced reactor is a 640-MWe PWR based on a concept developed by ABB. The PIUS reactor design is characterized by ABB as a concept, not a final design, because it is currently evolving in regards to significant details (Ref. 4-1). ABB submitted a PSID to describe the PIUS design and safety characteristics in 1990 (Ref. 4-2). Recently, ABB submitted a PSID Supplement to the NRC describing a number of important design changes (Ref. 4-1).

A primary ABB design objective was to make the probability of a core degradation accident extremely small. Reactivity is controlled by coolant boron concentration and the temperature of the reactor coolant. There are no mechanical control rods. The core is submerged in a large pool of highly borated water, and the core is in continuous communication with the pool water through pipe openings called density locks. There are no mechanical devices (e.g., valves) that separate the primary system and the pool. The primary coolant pumps are operated so that there is a hydraulic balance in the density locks (openings) between the primary coolant loop and the pool, keeping the pool water and primary coolant separated during normal operation.

An intrinsically passive scram function is provided for PIUS using only safety-grade equipment. The scram function is intrinsic in that it arises from the basic nature of (a) the effects of gravity, and (b) the physical properties of boron. The scram system is passive because the system is essentially self-contained or self-supported. It does not require pumped cooling water to perform its safety function. It relies on the natural force of gravity, the natural process of natural circulation, and does not use components that rotate or reciprocate (Ref. 4-3). The definition of passive systems in Ref. 4-3 states that passive safety systems may include features such as check valves, instruments, and single-action valves that satisfy specified conditions related to power supply. However, the PIUS intrinsically passive scram function includes no such valves or instruments.

Because the primary coolant and the heavily borated pool water are kept separated only when a pressure balance exists at the density locks between the primary coolant and the pool, and this pressure balance exists only when the primary coolant pumps are operating, any sufficiently large disruption of the pressure balance activates the lower and upper density locks. Highly borated water enters the primary system, usually through the lower density lock, mixes with any circulating primary coolant and enters the core resulting in a reduction in core power. If pool water continues to enter the primary through the lower plenum, a shutdown condition will be achieved. Upon even partial loss of forced circulation or excessive heatup of the primary coolant passing through the core, a natural-circulation flow will begin. For most event sequences, the natural-circulation path will be from the pool to the primary through the lower density lock and from the primary to the pool through the upper density lock. The pool is cooled by both active and intrinsically passive systems. Should the lower density lock be either inactive (due to operation of the control system or blockage), another natural-circulation path is available. Under such circumstances a natural-circulation path is available from the pool to the primary through the upper density lock and from the primary to the pool through the steam dome annulus at the top of the pool.

The thermal interface in the lower density lock is maintained by controlling the speed of the primary coolant pumps. However, the range of pump speed control is limited to 115% of

the nominal speed, and any disruption that cannot be controlled by a pump overspeed within that range will result in activation of the density locks. The normal reactor scram function reported in the original PSID submittal (Ref. 4-1) was accomplished by tripping one of the primary coolant pumps. Any unplanned loss of one or more primary coolant pumps (e.g., a loss of offsite power) will scram the reactor. In addition, if the reactor coolant overheats and the primary coolant pump speed limit is exceeded, the density locks will be activated and the core power will decrease. ABB has since modified the PIUS design to include a scram-valve system to provide the normal scram function discussed in the following paragraph.

As reported in the PSID Supplement (Ref. 4-1), ABB has changed the PIUS design so that a reactor scram is normally accomplished by opening valves in each of four lines connected between the reactor pool inside the reactor pressure vessel and the steam generator outlet plenum chamber on the suction side of the reactor coolant pumps. Highly borated water from the pool enters the primary coolant via forced circulation, and produces a reactor shutdown. Although the scram valves and associated piping are classified as safety grade, the primary coolant pumps are not. Successful operation of the scram-valve system requires primary coolant pump operation to create the driving potential that moves highly borated water from the pool to the primary system. Compared with the intrinsically passive scram function described in the preceding section, an important advantage of the scram-valve system is that fluctuations in coolant temperature and associated thermal-stress loadings on key components are reduced. However, the intrinsically passive and fully safety-grade scram function will function any time the scram-valve system fails to operate.

Safety-Related Design Goals

The safety-related design goals for the PIUS reactor are presented in Ref. 4-2 (p. 1.2-21). The PIUS reactor is designed to meet the following goals with respect to the safety of the public: The plant is to

- Meet the basic safety goals of the NRC;
- Be designed and constructed so as to reliably protect the core against damage in operation and accident situations and thus prevent severe accidents;
- Be designed and constructed so as to provide an extended interval after incidents and failures, before human action is required to prevent core damage;
- Be designed and constructed so as to require a minimum of operator action and equipment performance for reactor shutdown and core cooling after accidents;
- Be designed and constructed to limit the environmental releases following all accident scenarios to levels below those specified in 10CFR100;
- Be designed to meet the intent of NRC criteria for assessing the safety of advanced reactor concepts as described in SECY-88-203, "Key Licensing Issues Associated with Advanced Reactor Designs," particularly regarding the event categorization and the emergency planning considerations and the corresponding limitations of site boundary doses; and
- Have a high degree of resistance to malicious human intervention in terms of core damage.

Design Description

The following description of the PIUS concept is based upon the original PSID design (Ref. 4-2) as modified by the PSID supplemental information (Ref. 4-1). The design description is closely based upon these documents, but extensive use of quotes is avoided.

Key design data for the "PIUS 600" design are presented in Table 4-1. A schematic arrangement of the PIUS reactor is shown in Fig. 4-1.

PIUS is a four-loop PWR type reactor with a nominal core rating of 2000 MWt and 640 MWe. Although there are numerous similarities to current light water reactors, there are also significant differences. The reactor core is contained within a vertical cylindrical structure. The core and cylindrical structure, hereafter referred to as the reactor module, are submerged in a large pool of highly borated water (reactor pool). The reactor pool is enclosed by a prestressed, steel-lined concrete vessel, which also serves as the lower part of the reactor pressure vessel. The upper part of the reactor pressure vessel is the steel vessel extension, which is securely attached to the top of the prestressed concrete vessel. The normal forced-circulation path for the primary coolant is via four coolant loops, each having hot- and cold-leg piping that pass through the steel vessel extension and are connected to the reactor module near its top. Each primary coolant loop contains a steam generator and a primary coolant pump. The steam generators are located on the outside of the concrete reactor vessel. The primary coolant pumps are integrated into the lower steam generator plena. During normal (forced circulation) operation, the primary loop cooling water flows upward through the core, continues upward through the reactor module riser to the hot-leg piping, passes to the steam generator, passes through the primary coolant pump, returns to the top of the reactor module through the cold-leg piping, and passes downward through the downcomer to the bottom of the core to complete the coolant circuit.

The pressurizer is integrated into the top of the steel vessel extension. The reactor is pressurized by steam supplied from an electrically heated recirculation boiler that draws water from the pressurizer water volume at the top of the reactor pressure vessel. A pressurizer spray system is also provided. In the original PSID design, the pressurizer communicated with the primary loop via a funnel that formed the top of the reactor module and extended upward into the pressurizer steam space. Under normal operating conditions, the primary coolant communicated with the pressurizer only at liquid-steam interface inside the funnel. The funnel separated the primary coolant from the reactor pool present in the steel vessel extension, but on the outside of the funnel. The pressurizer steam space communicated with the reactor pool at the steam-liquid interface within the steel vessel extension. For the PSID Supplement design, the funnel diameter has been increased to coincide with the walls of the steel vessel extension, i.e., the primary coolant and the water pool in the pressurizer constitute a contiguous water mass. The connection between the reactor pool and the pressurizer now occurs through four pool stand pipes. That is, the reactor pool and the pressurizer steam space communicate within each of the four pool stand pipes. ABB states that this modification improves the hydraulic behavior of the loops during normal operation and transients.

A small diameter vertical pipe connects the pressurizer steam space to the reactor coolant within the reactor module near the top of the downcomer. This pipe is called the "siphon breaker." Somewhat below the level at which the cold legs attach to the reactor module, the bottom of the siphon breaker is open to the primary coolant flow in the downcomer. This siphon breaker arrangement effectively prevents draining the reactor primary loop and reactor pool in the event of cold-leg piping breaks.

The reactor module is open to the borated reactor pool at three locations. The lower density lock is located at the bottom of the reactor module. The upper density lock is located on the outer circumference of the reactor module at an elevation near the center of the reactor module. The third opening is through the funnel in the steel vessel extension. Normally the primary system water and the highly borated reactor pool water are not in direct communication as they are separated by the funnel wall and the pressurizer steam space.

However, during certain transient conditions, primary system water can overflow the top of the funnel and enter the reactor pool.

Each density lock consists of a bundle of vertical tubes that connect the water in the primary reactor module and the highly borated water in the reactor pool. There is no physical barrier to flow between the primary system and the reactor pool. Thus, a natural-circulation flow path between the reactor module and the large reactor pool is ever present and always available, thereby providing an intrinsically passive, safety-grade cooling system for the reactor core should the normal forced-circulation path through the four primary loops be interrupted for any reason. During normal operation, mixing of the cooling water in the reactor module and the highly borated water in the reactor pool must be prevented so that full-power operation of the core can be attained. This is accomplished by controlling the speed of primary coolant pumps to produce a pressure balance within the density locks between the liquids in the reactor pool and the reactor module. Use of small-diameter pipes as the fundamental component of the density lock promotes quiescent conditions in the density lock. Quiescent conditions are further promoted by ensuring that hot water exists on top of cold water in each density lock. The position of the interface between the hot primary-loop water and the cold reactor pool water is determined by temperature measurements in both the lower and upper density locks. The interface position information is used by the primary coolant pump control system to select a pump speed that will keep reactor pool water from entering the primary system.

There are no mechanical control rods in the PIUS reactor. Reactivity is controlled by coolant boron concentration and the temperature of the reactor coolant. Once stable conditions are created in the density locks, the boron concentration in the primary coolant can be controlled by introducing either fresh or highly borated water directly into the primary system coolant. However, a means of reliably introducing sufficient boron into the core to effect a reactor shutdown must also be provided. The intrinsically passive system for accomplishing the shutdown function relies on the always open lower and upper density locks. When sufficiently upset during transient conditions, such as loss of flow or a power mismatch, the interfaces in the density locks are interrupted and the highly borated water flows from the reactor pool into the core and shuts down the reactor. A natural-circulation flow path is then established from the borated pool through the lower density lock, through the core, and to the reactor pool through the upper density lock. Boron dilution is a relatively slow way of controlling reactor power. Therefore, the reactor core is designed to utilize the negative moderator temperature reactivity coefficient for rapid changes in power level.

Reactor shutdown by this intrinsically passive mechanism based upon flow through the lower and upper density locks is accompanied by some thermal cycling of key reactor components. Therefore, ABB has introduced a scram-valve system and intends that this scram-valve system will be the normal system used to scram the reactor. With the scram-valve system, reactor scram is accomplished by opening valves in each of four lines connected between the reactor pool inside the reactor pressure vessel and the steam generator outlet plenum chamber on the suction side of the reactor coolant pumps. Highly borated water from the pool enters the primary coolant via forced circulation and produces a reactor shutdown. Although the scram valves and associated piping are classified as safety grade, the primary coolant pumps are not. Successful operation of the scram-valve system requires primary coolant pump operation to create the driving potential that moves highly borated water from the pool to the primary system. Should the scram-valve system fail to operate at any time, the intrinsically passive and fully safety-grade scram function will automatically scram the reactor without any mechanical system actuations or operator action.

At the time a reactor scram is initiated, feedwater flow to the steam generator secondaries is terminated by ramping the flow to zero over an interval of about 20 s. The steam generator secondary inventory subsequently evaporates and the steam generators can no longer function as the heat sink. After steam generator secondary dryout, core-generated heat is deposited in the reactor pool. Any heat added to the reactor pool by the reactor core is removed by both active (nonsafety-grade) and passive (safety-grade) systems. The active system provides for forced circulation of pool water through heat exchangers and pumps outside the reactor vessel. The passive system is a closed loop utilizing coolers in the reactor pool and connecting piping to dry, natural draft cooling towers located on top of the reactor building. This latter system is intrinsically passive. Flow in this loop is by natural circulation. Each cooling loop is sized to provide sufficient cooling to prevent boiling of the reactor pool water in the event of a large LOCA. The natural draft cooling towers and heat exchangers are physically separated from each other by the reactor building superstructure in such a way that only one or two will be damaged by external hazards such as a impacting aircraft. In the extremely unlikely situation that both the active and passive pool cooling systems were lost, the core would continue to be cooled while the inventory of the reactor pool was boiled off, a period on the order of days.

The concrete vessel, reactor module, and primary coolant loops are all enclosed in a large containment structure of the pressure suppression type. All reactor equipment that normally operates at high pressure and temperature is located inside the containment. The containment is designed to withstand pressure and thermal loadings that would arise from a double-ended break of the largest pipe located in the containment. Steam entering the drywell is channeled to wetwell to be condensed in a large pool.

Key Safety Systems

The NRC has sponsored a study at Brookhaven National Laboratory (BNL) to perform a systems failure analysis of the PIUS plant systems that are used during normal reactor operation and postulated events (Ref. 4-4). One outcome of the BNL-study was an identification and listing of the major PIUS features (actions or systems) designed to mitigate an accident sequence. These actions or systems are briefly summarized in the following paragraphs based upon the information in Ref. 4-4. However, the BNL study was completed prior to the PIUS design modification that incorporated the scram-valve system. We have attempted to modify the BNL-descriptions to reflect this design change.

The PIUS system has five ways to achieve subcriticality. These are the active scram, manual scram, passive scram, feedwater reduction, and normal reactivity control using the chemical and volume control system. The rate of reactivity control differs substantially among the five.

1. ABB plans for the active scram mode to be based upon an active system, the scram-valve system. This is a rapid reactivity control system. The PIUS scram-valve system is a safety-grade system. The system consists of four scram loops, one for each reactor coolant loop, connected between the reactor pool inside the reactor pressure vessel steel extension and steam generator outlet plenum on the suction side of the reactor coolant pumps. Operation of the primary coolant pumps is required for the scram-valve system to operate. Activation of the lower and upper density locks is not required to scram the reactor. The primary coolant pumps are not safety-grade equipment.
2. The manual scram will also use the scram-valve system.

3. The passive scram in the original PSID design (Ref. 4-2) is still available for the PSID Supplement design (Ref. 4-1). This is a rapid reactivity control system. The passive system relies only on safety-class equipment. The passive scram system has been described in the overview and plant design portions of this section and the description will not be repeated here.

All three of the scram mechanisms circulate highly borated water between the pool and the primary, although portions of the pathway used in the passive scram differ from the pathway of the active and manual scram. Thus, there are mechanisms that could result in failure of the scram function. These are (1) low pool water, (2) insufficient boration of pool water, and (3) blocked flow. Instrumentation is included in the design to detect development of the first two situations. No instrumentation is included in the design to detect development of blockages in the scram-valve piping or the density locks. BNL concluded that it does not appear that sufficient water to disrupt the pool loop flow can be lost before the reactor will scram, which provides automatic protection against low-water caused failure (Ref. 4-4, p. D-2). BNL also concluded deboration of the pool is also unlikely (Ref. 4-4, p. D-2). BNL concluded that blockage of the lower density lock is the most credible means by which the passive scram system could fail (Ref. 4-4, p. D-3).

The BNL study concluded that the active, manual scram mechanisms in the original PSID design (Ref. 4-2) were neither independent nor diverse (Ref. 4-4, p. D-1). BNL believes that addition of the active or scram-valve system somewhat mitigates this conclusion (Ref. 4-4, p. 9-21). However, BNL still concludes that there are no truly diverse ways to scram the reactor as the source of boron in each case is the same pool.

4. Feedwater reduction can also be used as a means of reactivity control. A reduction in feedwater will cause the primary system coolant and fuel temperatures to increase. The PIUS core is designed such that an increase in primary temperature will result in the insertion of negative reactivity via the moderator temperature coefficient. Reactivity insertion upon primary temperature increase is a rapid mechanism.
5. Normal reactivity control in PIUS is managed with the reactivity control system (System 312). The amount of neutron absorber dissolved in the primary coolant is controlled. This is a relatively slow means of reactivity control because the flow rate of the reactivity control system is small relative to the volume of the primary loops.

In addition to the reactivity control systems and mechanisms, BNL identified a additional PIUS features (actions or systems) that are designed to mitigate an accident sequence.

1. The active pool cooling system consists of the reactor pool cooling system (System 323) and the nonsafety diesel-backed cooling systems (Systems 721 and 712) to the ultimate heat sink.
2. The passive pool cooling system is a safety-related system. In the event of loss of AC power, reactor pool coolers dissipate the heat to the ambient air via the passive closed cooling system (System 724) and dry natural draught cooling towers (System 714).
3. The closed cooling water system is the isolated heat exchanger system.

4. The service water system is a heat removal system that exposes the coolant to the atmosphere through the cooling towers.
5. The containment system (System 152) consists of the containment vessel, elevated cooling tower compartments, reactor service pool and fuel pool structure, containment vessel access penetrations, and containment vessel pipe penetrations. The functions of the containment are to contain radioactivity after accidents, contribute to the residual heat removal function, and to clean up the containment atmosphere.
6. The emergency water makeup system is provided to permit the addition of water to the reactor pool to prevent core uncover.

More detailed descriptions of these systems are found in the ABB-prepared documentation (Ref. 4-1 and 4-2 and in Ref. 4-4).

References

- 4-1. C. B. Brinkman, "PIUS PSID Supplemental Material (Contains Proprietary Information)," ABB Combustion Engineering Nuclear Power letter LD-93-020 to D. Scaletti (February 12, 1993).
- 4-2. "PIUS Preliminary Safety Information Document," ABB Atom (December 1989).
- 4-3. T. U. Marston, W. H. Layman, G. Bockhold Jr., and M. Montani, "Safety Criteria for the Passive Advanced Light Water Reactors," Proceedings of the International Conference on Design and Safety of Advanced Nuclear Power Plants, Tokyo, Japan (October 25-29, 1992).
- 4-4. R. Fullwood, P. Kroeger, J. Higgins, R. Youngblood, H. Cheng, G. Martinez-Guridi, G. Van Tuyle, W. Shier, R. Morante, and P. Soo, "Integrated Systems Analysis of the PIUS Reactor," Brookhaven National Laboratory document NUREG/CR-6111 and BNL-NUREG-25392 (October 1993).

Fig. 4-1.
PIUS PSID arrangement.

Table 4-1
PIUS 600 Key Design Data (Ref. 4-2, Table 1.2-1)

Core thermal power	2,000 MWt
Net electric power	640 MWe
Circulation water temperature	18°C (64°F)
Number of fuel assemblies	213
Core height (active)	2.50 m (8.2 ft)
Core equivalent diameter	3.76 m (12.3 ft)
Average fuel linear heat rate	11.9 kW/m (3.6 kW/ft)
Core inlet temperature	260°C (500°F)
Core outlet temperature	290°C (554°F)
Operating pressure	9 MPa (1,305 psi)
Core flow rate	13,000 kg/s (28,600 lbm/s)
Average burnup	45,000 MWd/t
Core vessel cavity diameter	12 m (40 ft)
Concrete vessel cavity volume	3,300 m ³ (870,000 gal.)
Concrete vessel total height	43 m (141 ft)
Concrete vessel thickness	7 m (23 ft)
Number of steam generators and coolant pumps	4

Section 5

Representative Operational and Accident Sequences

A summary description of the PIUS reactor and its key safety systems has been provided in Section 4. In this section, representative operational and accident sequences are considered and are described briefly. As the features of these sequences are explored and understood, the key systems, components, processes and phenomena can be established (Section 6). This information lays the foundation for evaluating the existing and planned PIUS data bases for completeness (See Sections 7 and 8).

The NRC has sponsored a systems failure analysis by BNL of the PIUS plant systems that are used during normal reactor operation and postulated events (Ref. 5-1). This study was based upon the original PSID design (Ref. 5-2) and did not include an evaluation of the scram-valve system incorporated into the PSID supplement design (Ref. 5-3). In addition, ABB Atom has completed a preliminary probabilistic safety analysis (PRA) study for PIUS (Ref. 5-4). This study considered the PSID Supplement design of Ref. 5-3). These studies provide the basis for a preliminary identification of representative operational and accident sequences.

The PIUS design is currently characterized by ABB as a design concept (Ref. 5-3). Therefore, it is not possible to confidently assess the frequency with which various operational and accident sequences will occur. Nevertheless, it is desirable to categorize accident sequences on the basis of frequency. Lacking quantified event trees and reliable frequency estimates, another approach was provided by the NRC to its contractors as the basis for event classification (Ref. 5-5). This approach has also been documented in Ref. 5-1. The approach assumes that the frequency of operational and accident sequence initiators, even for a concept design, can be estimated with an acceptable uncertainty. Initiator frequencies, together with the subsequent type and number of failures in systems responding to the event, are used to classify the total sequence into event categories (EC). The methodology is summarized in Fig. 5-1 from Ref. 5-5.

Based upon the information in Ref. 5-1, representative operational and accident sequences for event categories EC-I, EC-II, and EC-III are tabulated in Table 5-1. Some additional sequences are also included based upon the information in Ref. 5-4. This tabulation should not be considered to be complete. It is intended to be representative in that important sequences and phenomena are identified. However, Ref. 5-4 did not include frequency estimates for initiating events. Therefore, we have made a judgment about the appropriate event category for the initiating event and appropriate event category; we would look to revise these estimates later should work resume on PIUS certification efforts.

References

- 5-1. R. Fullwood, P. Kroeger, J. Higgins, R. Youngblood, H. Cheng, G. Martinez-Guridi, G. Van Tuyle, W. Shier, R. Morante, and P. Soo, "Integrated Systems Analysis of the PIUS Reactor," Brookhaven National Laboratory document NUREG/CR-6111 and BNL-NUREG-25392 (October 1993).
- 5-2. "PIUS Preliminary Safety Information Document," ABB Atom (December 1989).
- 5-3. C. B. Brinkman, "PIUS PSID Supplemental Material (Contains Proprietary Information)," ABB Combustion Engineering Nuclear Power letter LD-93-020 to D. Scaletti (February 12, 1993).

- 5-4. T. Pederson, "PIUS-Preliminary PRA Study, Analyses of draft 'Event Trees'," ABB Atom PROPRIETARY document A15 92-118, Revision 1 (January 22, 1993).
- 5-5. Z. R. Rosztoczy, "General Definitions of AOOs, DBAs, etc.," Office of Nuclear Regulatory Research, Division of Systems Research, Advanced Reactors Branch memorandum to J. E. Dyer (August 17, 1992).

Fig. 5-1.
NRC event category methodology.

Table 5-1
Representative PIUS Sequences
Reactor Scram

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Reactor scram	Scram-signal initiator	Failure of scram valves to open or Pool boron concentration at allowed minimum of 1800 ppm	Pool boron concentration below allowed minimum or Blockage of lower density lock

Table 5-1 (continued)
Representative PIUS Sequences
Loss of Reactor Coolant Pump

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Loss of reactor coolant pump (LCP)	LCP initiator and scram-valve actuation (Ref. 5-d, p. 8)*	Failure of scram valves to open or Pool boron concentration at allowed minimum of 1800 ppm	Pool boron concentration below allowed minimum or Lower density lock blockage

*Event identified in reference but no information about the initiator frequency was provided. Assignment to EC-I is author's choice.

Table 5-1 (continued)
Representative PIUS Sequences
Loss of Offsite Power

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Loss of offsite power (LOSP)	LOSP initiator (Ref. 5-a, p. 7-24)	Pool boron concentration at allowed minimum of 1800 ppm	Failure of active pool cooling (Ref. 5-a, p. 7-24) or Lower density lock blockage or Pool boron concentration below allowed minimum

Table 5-1 (continued)
Representative PIUS Sequences
Turbine Trip

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Turbine trip (TT)	TT Initiator (Ref. 5-a, p. 7-27) or TT initiator and failure of active scram (Ref. 5-a, p. 7-27) or TT initiator and failure of steam bypass valve to open (Ref. 5-a, p. 7-27)	Pool boron concentration at allowed minimum of 1800 ppm	Failure of active pool cooling or Failure of active and manual scrams or Failure of steam bypass valve to open concurrent with failure of active pool cooling or Failure of steam bypass valve to open concurrent with failure of active scram or Failure of steam bypass valve to open concurrent with failure to activate steam generator and safety relief valves (Ref. 5-a, p. 7-27, all above) or Lower density lock blockage or Pool boron concentration below allowed minimum

Table 5-1 (continued)
Representative PIUS Sequences
Feedwater Transient

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Feedwater transient (FWT)	FWT initiator (Ref. 5-a, p. 7-37) or FWT initiator and failure of active scram (Ref. 5-a, p. 7-37)	Pool boron concentration at allowed minimum of 1800 ppm	Failure of active pool cooling (Ref. 5-a, p. 7-37) or Failure of active and manual scrams (Ref. 5-a, p. 7-37) or Lower density lock blockage or Pool boron concentration below allowed minimum

Table 5-1 (continued)
Representative PIUS Sequences
Primary Loop Boron Dilution

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Primary loop boron dilution (PLBD)	PLBD initiator (Ref. 5-d, p. 12)* PLBD initiator and failure of active scram	Pool boron concentration at allowed minimum of 1800 ppm	Failure of active and manual scrams or Lower density lock blockage or Pool boron concentration below allowed minimum

*Event identified in reference but no information about the initiator frequency was provided. Assignment to EC-I is author's choice.

Table 5-1 (continued)
Representative PIUS Sequences
Loss of Reactor Pool Cooling

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Loss of reactor pool cooling (LRPC)	LRPC initiator (Ref. 5-d, p. 14)* LRPC initiator and failure of active scram	Pool boron concentration at allowed minimum of 1800 ppm	Failure of active and manual scrams or Lower density lock blockage or Pool boron concentration below allowed minimum

*Event identified in reference but no information about the initiator frequency was provided. Assignment to EC-I is author's choice.

Table 5-1 (continued)
Representative PIUS Sequences
Small Primary Pipe Break

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Small primary pipe break (SPPB) Locations to consider: hot-leg cold-leg steam generator (Ref. 5-d)	Not Applicable	SPPB initiator (Ref. 5-a, p. 7-22) or SPPB initiator and failure of active scram (Ref. 5-a, p. 7-22)	Failure of active and manual scrams (Ref. 5-a, p. 7-22) or Pool boron concentration at allowed minimum of 1800 ppm or Lower density lock blockage or Pool boron concentration below allowed minimum

Table 5-1 (continued)
Representative PIUS Sequences
Steamline Break Inside Containment

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Steamline break inside containment (SBIC)	Not applicable	SBIC initiator (Ref. 5-a, p. 7-31) SBIC initiator and failure to isolate ends of break (Ref. 5-a, p. 7-31)	Failure of active scram (Ref. 5-a, p. 7-31) or Failure of active scram and concurrent failure to isolate ends of break (Ref. 5-a, p. 7-31) or Failure of active and manual scrams, concurrent failure to isolate ends of break, and containment failure (Ref. 5-a, p. 7-31) or Pool boron concentration at allowed minimum of 1800 ppm

Table 5-1 (continued)
Representative PIUS Sequences
Steamline Break Outside Containment

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Steamline break outside containment (SBOC)	Not applicable	SBOC initiator (Ref. 5-a, p. 7-33) SBOC initiator and failure to isolate ends of break (Ref. 5-a, p. 7-33)	Failure of active scram (Ref. 5-a, p. 7-33) or Pool boron concentration at allowed minimum of 1800 ppm

Table 5-1 (continued)
Representative PIUS Sequences
Large Primary Pipe Break

Event Title	EC-I Anticipated Operational Occurrences	EC-II Design Basis Accidents	EC-III Severe Accidents
Large primary pipe break (LPPB) Locations to consider: hot leg cold leg (Ref. 5-d)	Not applicable	Not applicable	LPPB initiator (Ref. 5-a, p. 7-19) LPPB initiator and failure of active scram (Ref. 5-a, p. 7-19) or LPPB initiator and pool boron concentration at allowed minimum of 1800 ppm

Section 6

Key Systems, Components, Processes, and Phenomena

PIUS is currently characterized by ABB as a design concept (Ref. 6-1). Therefore, knowledge of the key systems, components, processes, and phenomena is evolving and uncertainties exist. Within this context, it is still important to make an effort based upon current knowledge to characterize the key PIUS systems, components, processes, and phenomena. Those systems, components, processes, and phenomena that are most important to the safety performance of the PIUS reactor and least understood, are most in need of testing support. Thus, once developed, this information provides an important basis for assessing both the adequacy and priorities of planned and completed PIUS test programs (Section 7).

There are several different approaches to identifying potentially important systems, components, processes, and phenomena in the PIUS reactor. One approach is to examine the results of early safety reviews of other advanced reactors conducted by the NRC and identify key systems, components, processes, and phenomena in those reactors that have a counterpart in the PIUS reactor. A second approach is to examine the PIUS-specific analytical information developed by Los Alamos as part of its NRC-funded thermal-hydraulic systems and safety studies within FIN L2447, Thermal-Hydraulic Assessment of PIUS. A third approach is to examine the PIUS-specific analytical and experimental information developed by ABB. We have found the first approach to be a rich source of information because many of the review activities for other advanced reactor concepts have identified and highlighted information about key systems, components, processes and phenomena. The Los Alamos studies have also proven helpful because there has been an early and continuing focus on identifying this information. The ABB-prepared information is rich in its descriptions of experimental and analytical results, but ABB has not highlighted the needed information regarding key systems, components, processes, and phenomena. Thus, ABB has not, to our knowledge, summarized and documented its extensive insights in a manner that highlights the desired information. Some insights into the importance assigned to various systems, components, processes, and phenomena can be inferred from the focus placed by ABB on analysis and experimental efforts.

In the following sections, the information regarding key systems, components, processes, and phenomena in each of the three areas will be examined. At the conclusion of this section, a single table summarizing the findings regarding key systems, components, and phenomena will be presented.

Ideally, an effort should be made to rank the systems, components, processes, and phenomena in order of importance. One formal approach for identifying and ranking key systems, components, processes, and phenomena is to develop a phenomena identification and ranking table (PIRT). This is a large effort, particularly if a number of scenarios are examined. PIRTs have been developed for several light water reactor applications (e.g., Ref. 6-2), a Savannah River reactor (Ref. 6-3), and the New Production Heavy Water Reactor (Ref. 6-4). Although such an extensive effort is not warranted for the PIUS preapplication review, development of a PIRT for PIUS may warrant consideration if the reactor is submitted to the NRC for design certification.

NRC Advanced Passive Reactor Reviews

A primary focus of NRC reviews to date of the advanced reactors has been the passive, safety-grade cooling *systems* (Ref. 6-5, Ref. 6-5, Enclosures 1 and 2; Ref. 6-6; and Ref. 6-7, Enclosures 1 and 2). *Key components* in the passive cooling systems are the core, the flow enclosing structures such as pipes and vessel structures, and the interconnections with other safety and nonsafety-class systems. The *key processes* are natural circulation to remove decay heat at high and low pressures; reliance on small differences in temperature, pressure, or elevation for operation; thermal-hydraulic interactions with other safety and nonsafety systems; interactions with pumped systems that may operate concurrently; establishment of a long-term, stable recirculatory cooling mode; flow redistribution; flow maldistribution; core uncover; oscillatory neutronic feedback; and boron transport under gravity-driven conditions (natural circulation). *Key phenomena* are buoyancy-induced convection; flow and heat transfer in low-flow regimes; flow and heat transfer in the presence of noncondensable gases; two-phase flows; two-phase pressure losses, countercurrent flow; flow blockages; flow past obstructions; delayed or intermittent boron injection; boron streaming; boron diffusion; and manometer oscillations.

Several other *systems* have been identified that have a relationship to the PIUS reactor. When there is only a single source of safety-grade water, then level phenomena are of interest to the NRC reviewers (Ref. 6-5, Enclosure 2). Similarly, the existence of a single passive residual heat removal system is of interest if there are multiple flow paths and the potential for systems interactions exists (Ref. 6-5, Enclosure 2). Complex flow networks are of interest because unforeseen flow distributions or maldistributions might arise (Ref. 6-6).

Few *components* have been the focus of special attention. However, NRC reviewers have focused attention on valves, particularly when failure of the valves to function will degrade safety system performance (process) or when the valves are expected to function because small pressure differences exist (Ref. 6-5, Enclosure 2).

Several *processes* have been identified for special attention. These include systems interactions (Ref. 6-5, Enclosure 1); boron injection, transport, diffusion, and streaming (Ref. 6-5, Enclosure 2); oscillatory neutronic feedbacks (Ref. 6-7, Enclosure 2); interactions between connected pools (Ref. 6-7, Enclosure 2); and establishment of a long-term, stable, recirculatory cooling mode (Ref. 6-5, Enclosure 2).

Although the NRC reviews of the PIUS reactor were few and of a preliminary nature, some key systems, components, processes, and phenomena were identified. The reactor pool cooling system and several related components, the pool coolers and the upper density lock, were identified (Ref. 6-8). The key process of interest was mixing of liquid exiting the upper density lock into the pool. The key phenomena of interest was streaming. The flow structures, particularly those in the lower reactor module, have also been identified as a system of interest (Ref. 6-8). Several of the components of interest in this system are the gas baffle and the pool water inlet tubes. The process of interest is boron transport from the reactor pool to the core.

Several PIUS *components* have been identified for special attention. The siphon breaker is activated during vessel blowdown at high rates (process), and the phenomena of interest is flow interruption (Ref. 6-8). The density locks are unique to the PIUS design and are the pathway for introduction of highly borated water into the primary system (Ref. 6-9). Because PIUS contains no control rod system, reliable and predictable operation of the density locks and the associated processes and phenomena are of great importance.

Los Alamos Perspectives

An early Los Alamos effort was to prepare a list of PIUS systems and components, light water reactor counterparts, key characteristics of the listed systems and components, and other relevant factors (Ref. 6-10). During the course of the "PIUS Transient Analysis Effort," Fin L2447, the PIUS reactor was extensively analyzed. Both multidimensional and fully one-dimensional models of the PIUS reactor were prepared, and analyses were performed for the pump-trip scram and main steam line break sequences (Refs. 6-11, 6-12, 6-13, 6-14, 6-15). Subsequently, ABB modified the design and submitted these changes as the PSID Supplement (Ref. 6-16). Los Alamos revised the multidimensional and one-dimensional TRAC models to reflect the PSID Supplement design and performed a number of sequence analyses and sensitivity studies.

The representative operational and accident sequences analyzed were (1) reactor scram with the scram-valve system, (2) loss of offsite power, (3) main steam line break, (4) small-break loss of coolant, and (5) large-break loss of coolant. In addition, a number of sensitivity studies were completed examining the effect of pool boron concentration, blockages of the lower density lock, pump speed, etc. These studies are documented in Refs. 6-17 through 6-21. With the insights derived from these analyses and related insights from the BNL study (Ref. 6-22), the tabulation of systems, components, key characteristics and factors was updated and restructured to the format of systems, components, processes, and phenomena (Table 6-1). Not all systems identified in the BNL study (Ref. 6-22) are presented in Table 6-1. Systems such as the Plant and Turbine Control Systems and Electric Power are not included because the impact of failures in these systems are usually modeled as initial conditions to the analyses performed in support of FIN L2447.

Judgments about which systems, components, processes, and phenomena are most important (key) in defining the safety characteristics of the PIUS reactor are not provided in Table 6-1. This unprioritized list is intended to be a general source listing of PIUS systems, components, processes, and phenomena that have an impact on the thermal-hydraulic and systems safety characteristics of the PIUS reactor. This list can be refined should the review effort for PIUS continue at a future time. More detailed descriptions of the systems and components are found in Refs. 6-16, 6-22 and 6-23. Some descriptions of the systems and components in Table 6-1 may be inconsistent with the current ABB design. The PSID Supplement (Ref. 6-16) documented modifications and changes to the design but did not detail the impact of these changes on each of the PIUS systems.

Los Alamos perspectives on the key PIUS safety-related systems, components, processes, and phenomena will be presented at the end of this section.

ABB Perspectives

As stated in the introductory comments to this section, ABB has not explicitly summarized its views on the key PIUS systems, components, processes, and phenomena. However, some insights can be obtained by examining the PSID document (Ref. 6-23) and reviewing the safety-related design and test programs that are emphasized therein and in related documentation (Ref. 6-24).

The importance of the "inherent boron shutdown system" and "residual heat removal systems" are emphasized in the PSID. The importance of these system is emphasized in the discussion of previously completed and planned integral test programs. The proof-of-principle integral test was conducted in the Magne loop. A small-scale integral test was conducted in the ATLE test loop. Further tests are planned in a modified ATLE test loop

that will more closely represent PIUS; the test program will investigate more severe accident sequences such as the large-break loss-of-coolant accident. A larger scale integral test facility is envisioned for the future. Each of integral facilities, present or planned, has either tested or will focus on the inherent boron shutdown system and the residual heat removal system. In addition, tests are envisioned for the passive pool decay heat removal system, flow distribution in the riser at the connection to the upper density lock, and transient flow conditions in the reactor pool.

Testing programs have either been conducted or are planned for new components in these systems. Testing has focused on those components that play a key role in the inherent boron shutdown and residual heat removal systems for which the experience base is limited. These components are the prestressed concrete reactor vessel (PCRVR), lower density lock, and upper density lock. Future tests are planned in addition to the test completed to date. ABB also notes that the core is a key component and that it has been designed to have a low power density, low linear heat flux, and negative moderator and fuel reactivity coefficients. Data required for verification of the actual PIUS core design during operation, and transients, as well as accidents, will be generated during a test program.

The key processes associated with the inherent shutdown and residual heat removal systems are natural circulation of highly borated water from the reactor pool into the primary through the lower density lock, passage through the core, and return to the reactor pool through the upper density lock. In the natural-circulation mode, decay heat is deposited in the reactor pool. A passive, safety-class system is provided to cool the reactor pool. The large volume of the reactor pool and the incorporation of the siphon breaker are intended to keep the core covered and cooled for extended periods of time in the unlikely event that nonsafety grade active cooling systems and the passive pool cooling systems fail.

Since the inherent boron shutdown and residual heat removal systems envision flows of highly borated cooling water from the reactor pool into the primary, the density locks must always remain open. Therefore, one phenomena highlighted by ABB is blockage of the density locks.

Summary

In the previous subsections of Section 5, insights arising from NRC advanced passive reactor reviews, Los Alamos perspectives, and ABB perspectives have been discussed. An extensive list of PIUS systems, components, processes, and phenomena has been presented in Table 6-1. Some identification of the more important systems, components, processes, and phenomena is desirable. Stated another way, those systems, components, processes, and phenomena that are most important to the safety performance of the PIUS reactor and least understood, are most in need of testing support.

In Section 2, perspectives about data base requirements and needs were identified based upon a licensing certification perspective. The advanced light water reactors, of which PIUS is one variation, have replaced the well understood active safety-class systems with passive systems that do not have a proven performance history. These systems rely on small differences in temperature, pressure, or elevation for operation and these may be affected by relatively minor variations in these parameters. Likewise, interactions between the passive safety-class systems and other nonsafety systems may be of more importance. Testing, both separate effect testing of individual components and systems and integral testing of systems, is the preferred mode of developing the needed understanding about the performance characteristics of the advanced light water reactors.

In Section 3, perspectives about data base requirements and needs were identified by considering the need for validated computational tools (code) to predict the behavior of the advanced light water reactors during representative operational and accident sequences. Code adequacy is demonstrated in a two-stage process. The adequacy of the individual building blocks of the code is confirmed after determining how the individual models and correlations display the requisite pedigree, applicability, and fidelity. Successful integration of the building block models and correlations is confirmed after determining how the global code features display the requisite applicability and fidelity.

In Section 5, representative operational and accident sequences were identified. This set of sequences, or a similar set, will be used to certify the safety performance of the PIUS reactor. An examination of these sequences provides insights into the conditions under which the PIUS systems and components are required to operate.

Given this background, a definition of the key PIUS systems, components, processes, and phenomena is attempted. In this context, the concept of "key" has several component parts. First, the systems, components, processes, and phenomena define and characterize the safety-related behavior of the PIUS reactor. Second, the systems, components, processes, and phenomena are unique in either concept or application within the PIUS system. Third, important systems interactions are possible. Fourth, an existing, applicable, and adequate testing data base does not exist. Applying these concepts is an exercise in judgment. This is a first attempt and the author anticipates that significant revisions will occur to both the definition of what is "key" and the resultant identification of systems, components, processes, and phenomena as more complete knowledge about the PIUS reactor evolves through subsequent analysis and testing activities. The key PIUS systems, components, processes, and phenomena are summarized in Table 6-2.

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Table 6-1

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Control systems (See Refs. y2, y1, and x7)		No No Yes No Yes Yes Yes Yes	Lower density lock control Upper density lock control Flow control Reactivity control Pressure control Temperature control Feedwater and turbine control Signal generation and processing	Thermal interface movement Boron transport Energy transport Flow Signal delays
Plant protection systems (See Refs. y2, y1, and x7)		Yes No Yes Yes Yes Yes Yes Yes Yes Yes Yes	Trip on manual Trip on low pool boron concentration Trip on low pressurizer pool level Trip on high pressurizer pool level Trip on low pressurizer pressure Trip on high pressurizer pressure Trip on high neutron flux increase rate Trip on high neutron flux level rate Trip on high core exit coolant temperature Trip on high differential pressure across steam generator outlet nozzle Trip on high containment pressure	Low boron concentration Low reactor pool level High reactor pool level Low primary pressure High primary pressure High rate of core power increase High core power High coolant temperature Blowdown Primary blowdown

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Reactor internals, fuel assemblies, and primary recirculation system	Reactor internals including		Pumped-circulation coolant flow Natural-circulation coolant flow	Temperature differences Pressure differences
	Core envelope structure	No	Density lock activation Boron transport and diffusion	Level differences and overflows Density differences
	Core bottom and side support structures	Yes	Flow distribution-multiple paths Mixing flows-boron concentration	Boron concentration variations Flow reversal
	Reactor inlet plenum components and lower density lock	No	Flow regime changes affecting Fluid-surface heat transfer Density and buoyancy Changes in liquid levels Nitrogen injection and extraction	Flow stagnation Flow resistance due to structures and geometry Flow oscillations Flow swirling at crossovers Condensation Evaporation Thermal stratification Breaking of siphon Dimensional variations in above (axial, radial, and azimuthal)
	Flow guide structures Numbers 1-4 (includes flow shiftover region, upper density lock, siphon breaker and main coolant nozzle connections)	No		

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Reactor internals, core, and primary recirculation system	Fuel assemblies Core	Yes	Pumped-circulation coolant flow Natural-circulation coolant flow Boron transport and diffusion Reactivity insertions via boron, moderator temperature, void, and fuel temperatures Mixing flows-boron concentration Flow regime changes affecting Fluid-surface heat transfer Density and buoyancy	Temperature differences Pressure differences Level differences Core uncover and recovery Buoyancy driven flows Boron concentration variations Reactivity changes Decay heat Power peaking (axial, radial) Fuel rod stored energy Gap conductance Flow reversal Flow stagnation Flow resistance due to structures and geometry Flow oscillations Flow path blockage Crud buildup Condensation Evaporation Dimensional variations in above (axial, radial, and azimuthal)

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Reactor internals, core, and primary recirculation system	Primary recirculation system including		Pumped-circulation coolant flow Single phase Two phase Pump coastdown	Temperature differences Pressure differences Level differences Density differences
	Hot- and cold-leg piping	Yes	Natural-circulation coolant flow Boron transport and diffusion	Boron concentration variations Flow reversal
	Steam generators	Yes	Flow distribution-multiple paths Mixing flows-boron concentrations	Flow stagnation Flow resistance due to structures and geometry
	Main coolant pumps	Yes	Flow regime changes affecting Fluid-surface heat transfer Density and heads Changes in liquid levels Break flows	Flow oscillations Condensation Evaporation Noncondensable trapping
Scram-valve system	scram valves piping	Yes	Pumped flow Flow distribution-multiple paths Mixing flows-boron concentrations	Flow regime Flow resistance Flow oscillations
PCRv and reactor pool	PCRv Highly borated water pool	Yes* (gas-cooled reactors) No	Natural-circulation flows through lower and upper density locks Heat addition to pool (flows, conduction from structures) Heat absorption from pool (pool coolers, concrete cooling) Heat and mass exchange at pressurizer interface	Pool temperature Pool height Pool boron concentration Flow streaming Mixing Level oscillations Primary to pool spillover

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Pressurizer system	Anchor flange Cylindrical vessel steel extension Vessel steel head	Yes	Steam addition Steam extraction Water spray Noncondensable gas removal Heat and mass exchange at reactor pool interface	Pressure variation Level (volume) variation Steam condensation Water evaporation Mixing
Reactivity control	Filtered water supply system Distilled water supply system Borated water supply system Inlet-Outlet system Normal withdrawal system Temporary discharge system	No	Boron addition Shutdown to hot or cold conditions Boron dilution Reactor startup Compensate for fuel burn up Interface level control in the upper density lock	Boron concentration Mixing

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Pressure relief system	Safety valves Relief valves Discharge pipe system Emergency blowdown valves	Yes	Steam discharge Two-phase discharge Water discharge	Pressure decrease Loss of primary fluid
Condensation system	Pool Blowdown pipes Valves Instrumentation	Yes	Receive steam and two-phase flows Air flow Noncondensable gas flow Air and noncondensable gas discharge to the drywell Pool heatup Pool pressurization Water discharge	Condensation Mixing Air and gas percolation through pool

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Containment system	Containment vessel	Yes	Receive steam and two-phase flows	Pressure Temperature
	Elevated cooling tower compartments		Air flow	
	Reactor service pool and fuel pool structure		Noncondensable gas flow	
	Containment vessel access penetrations		Air and noncondensable gas discharge to the drywell	
	Containment vessel pipe penetrations			

Table 6-1 (Continued)

PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Cooling systems (safety-class) including Reactor pool cooling tower Passive closed cooling system for reactor pool	Cooling tower Water coolers Piping Heat exchanger Inner containment isolation valves Outer containment isolation valves Pressurizing vessel	Yes	Natural-circulation flow	Pressure Temperature

Notes:

1. The component or system is unique to PIUS, but some LWRs may have similar features.
2. No effort has been made to identify dominant phenomena.
3. This factor is not unique to PIUS, but accurate modeling may be more important for PIUS because the safety-related design characteristics of PIUS are strongly influenced by this factor.

Table 6-2

Key PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Control		No	Density lock control	Thermal interface movement
Plant Protection		Yes	Measure pool boron concentration	Low boron concentration
Reactor internals, fuel assemblies, and primary recirculation	Reactor internals including core envelope, siphon breaker, density locks, core support and flow guide structures	No	Pumped-circulation coolant flow Natural-circulation coolant flow Activation of density locks Boron transport and diffusion Flow distribution-multiple paths Mixing flows-boron concentration	Flow oscillations Thermal stratification Breaking of siphon Dimensional variations in above (axial, radial, and azimuthal)
(continued)	Fuel assemblies Core	Yes	Pumped-circulation coolant flow Natural-circulation coolant flow Boron transport and diffusion Reactivity insertions via boron, moderator temperature, void, and fuel temperatures Mixing flows-boron concentration Flow regime changes affecting Surface-fluid heat transfer Density and buoyancy	Core uncover and recovery Buoyancy driven flows Reactivity changes Power peaking (axial, radial) Flow resistances Flow oscillations Flow path blockage Dimensional variations in above (axial, radial, and azimuthal)
(continued)	Primary recirculation system including piping, steam generators, and coolant pumps	Yes	Pumped-circulation coolant flow Pump coastdown Flow distribution-multiple paths Mixing flows-boron concentrations Break flows	Flow regime Flow resistance Flow oscillations

Table 6-2 (Continued)

Key PIUS Systems, Components, Processes, and Phenomena

PIUS System	PIUS Component	LWR Counterpart	Process	Phenomena
Scram-valve	Scram valves piping	Yes	Pumped flow Flow distribution—multiple paths Mixing flows-boron concentrations	Flow regime Flow resistance Flow oscillations
PCRVR and reactor pool	PCRVR and highly borated water pool	Yes* No	Natural-circulation flows through lower and upper density locks Heat addition to pool Pool Cooling	Pool temperature Pool boron concentration Thermal and boron mixing Level oscillations
Pressurizer system	Hardware	Yes	None	None
Reactivity control	Hardware	No	None	None
Pressure relief system	Hardware	Yes	None	None
Condensation system	Pool and hardware	Yes	None	None
Containment system	Containment vessel and other structures	Yes	None	None
Pool cooling (safety class)	Hardware	Yes	Natural circulation flow	Pressure Temperature

* Data base exists for gas-cooled reactors.

Section 7

Assessment of Existing and Planned Data Bases

The primary objective of this document is to assess the data bases that have been used by ABB to validate the "steady-state and transient neutronic and thermal-hydraulic computer codes" and to review other relevant data bases that may exist. The combined data bases are assessed with regard to "their quality and completeness for validating code features *important* to PIUS performance." Our assessment of the quality and completeness (adequacy) of existing and planned data bases is provided in this section. A logical foundation has been provided in previous sections of this report to support the adequacy assessments, which are of necessity qualitative judgments.

Earlier sections in this report have identified code features *important* to PIUS performance (Section 3). These code features are directly related to the reactor design (Section 4), representative operational and accident sequences for the design (Section 5), and the phenomena that would occur during PIUS operational and accident sequences (Section 6). In addition, data base needs arising from NRC precertification and certification reviews of other advanced passive reactor concepts have been identified (Section 2), because the precedents arising from these reviews provide important insights about current NRC data base adequacy standards.

The adequacy assessments in this section, although based on a logical structure, require qualitative judgments regarding importance, completeness, and adequacy of data base work completed and planned. ABB plans to do additional testing. The design may change as it matures beyond the conceptual design stage. Should the PIUS reactor proceed to the certification stage in the United States, the conclusions of this report should be revisited if additional design and test information are available.

Data Base Assessment Perspectives

Several perspective statements are provided about the assessments that follow.

First, with the exceptions noted in the following statements, essentially all the *directly* applicable data bases have been generated by ABB. Little *directly* applicable data base information from other sources exists.

Second, some of the PIUS components are similar to those in other commercial reactors, e.g., steam generators, reactor coolant pumps, and valves. For such components, existing performance data should be adequate if the PIUS components are sufficiently similar. This is particularly true if the components are not essential to the safety-related performance of systems. However, the performance data should be made available to the NRC to support its program of independent safety assessment.

Third, ABB has conducted or participated in several experimental programs for components that are unique to PIUS, e.g., the density locks and wet thermal insulation. However, only brief descriptions of these tests appear in the documentation provided to the NRC and Los Alamos. The test reports have not been reviewed by either the NRC or Los Alamos.

Fourth, proof-of-principle integral tests have been conducted by both ABB and the Japanese (see Table 7-3). In addition, ABB has constructed and operated a small-scale PIUS integral test facility (see Table 7-3). ABB provided a detailed facility description, facility drawings, and test data to the NRC and Los Alamos. The proof-of-principle and small-scale integral test facilities are tall, narrow facilities; they are essentially one dimensional. No data base currently exists for assessing multidimensional processes and phenomena in PIUS.

Fifth, the dominant reactivity insertion mechanisms in PIUS are those associated with introducing boron into the core and the moderator temperature coefficient. PIUS does not contain a mechanical control and shutdown system. No testing program, either previously completed or currently envisioned by ABB, *directly* simulates these key processes, e.g., reactor shutdown by boron.

Sixth, the separate effect and integral test programs must be considered as a whole. It is not possible to determine and assert with absolute certainty that a particular separate effect tests and particular integral test programs must be conducted. For example, a comprehensive separate effect test program may deliver sufficient data about key systems, components, processes and phenomena to permit a simpler integral test facility design than would be the case if the comprehensive separate effect test data did not exist. The converse could also be true.

Seventh, the existing PIUS data base is small. ABB has described additional testing that is planned should the PIUS design proceed to the detailed design phase. While clearly identifying those experiments have been completed and those experimental programs that are planned in the following assessments in this section (see Tables 7-1 through 7-4), the data base adequacy assessments are based upon the total test program, completed and planned, as described by ABB. Should ABB redefine its testing program in the future, the present assessment of the data base adequacy should be revisited.

Eighth, test documentation and results have been reviewed for only one separate effect or integral test program, the small-scale integral tests conducted in the ATLE facility. Test design and result documentation were not requested from ABB and, therefore, were not reviewed for the remainder of the separate effect and integral test programs. Only sketchy outlines exist for planned programs. This important factor must be considered when evaluating the conclusions presented in this section.

The evaluations are divided into several categories. Separate effect and key process tests focus on processes, individual components, and some systems. Separate effect and key process tests conducted to date are summarized in Table 7-1. Planned separate effect and key process tests are summarized in Table 7-2. Integral tests focus on entire safety-systems and total-system performance. Integral tests conducted to date are summarized in Table 7-3. Planned integral tests are summarized in Table 7-4.

Assessment of Separate Effect and Key Process Tests Data Base

Separate-effect tests focus on phenomena, processes, components, or even systems. The separate-effect tests conducted by ABB to date focus on several components: the PCRV

(Ref. 7-1), density locks (Ref. 7-1), and wet thermal insulation (Ref. 7-1). The PCRV and density locks were identified as key components in Table 6-2. The wet thermal insulation was not so identified because its function appears to be more performance than safety related. Although ABB is considering additional testing related to the construction of the PCRV, the existing data base for PCRVs is extensive. The density lock tests conducted to date have explored basic processes and phenomena in the density lock component. These include the thermal stability of the interface, characterizing the rate of boron diffusion across a thermally stratified layer, and the effect of fluid "sloshing" on the interface.

ABB plans additional separate effect tests that focus on phenomena, processes, components, and systems. Tests will be conducted in an approximately 1/10 scale test to characterize the mixing process as hot water released through the upper density lock into the pool mixes with the cold water in the reactor pool (Refs. 7-2, 7-3). A full-scale test of "sloshing" in density lock pipes exposed to representative turbulence conditions is planned to study the stability of thermally stratified layers (Ref. 7-1). A full-scale density lock module will also be constructed to characterize the transport mechanisms in the density lock, verify design features provided to reduce excessive mixing of primary coolant and pool water in the density lock, and characterize the dynamic behavior of the density lock (Refs. 7-1, 7-3). Tests will be conducted in a reduced-scale facility that models the upper density lock and a sufficient portion of the riser to measure the flow conditions in the riser and characterize the disturbance to the density lock from the reactor coolant flow (Ref. 7-2, 7-3). A full-scale test of the siphon breaker during conditions representative of normal operating conditions, startup and shutdown, and primary system cold-leg breaks are planned (Refs. 7-1, 7-2, 7-3). A test program will be conducted to verify the departure from nuclear boiling (DNB) correlation used in the analysis of normal operation, transients, and accidents (Ref. 7-3). Component tests include the wet thermal insulation (Refs. 7-1, 7-3), PCRV (Ref. 7-3), steam generator (Ref. 7-1), and reactor coolant pump (Ref. 7-1). Of these tests, the reactor coolant pump tests are deemed most important to developing data that can be used for preparing code models that will be used for safety analysis.

Based upon the assessment of key systems, components, processes, and phenomena identified in Table 6-2, some additional testing appears to be desirable. *First*, a demonstration of the system used to measure pool boron concentration is needed. If well-proven hardware systems currently exist for the measurement of boron concentration under prototypic conditions, the demonstration may consist of certifying the system adequacy and applicability via documentation. *Second*, characterization of flow distributions in the lower density lock, startup gas lock baffle, natural-circulation plenum, and core appear to be desirable. These tests are needed to define the potential for flow streaming as highly borated water enters the primary system from the reactor pool. A related phenomena of interest is the mixing of lightly borated primary system water passing through the downcomer into the core and highly borated water entering the primary through the lower density lock. This information can also be collected in an appropriately designed integral test and ABB has identified a large-scale tests that may satisfy this need. *Third*, testing is needed to characterize reactivity control with boron and perhaps moderator temperature coefficient (MTC). The requirements of 10CFR52.47(b)(2)(i)(A) mandate, in the absence of a prototype, the requirements for certification of a standard design that utilizes simplified, inherent, passive, or other innovative means to accomplish safety functions. The demonstration required of the applicant of these features can be by either analysis, appropriate test programs, experience, or a combination thereof. Therefore, ABB may choose to demonstrate the reactor shutdown mechanisms by means other than testing, e.g., analysis. If ABB chooses this approach, it will be necessary to document and defend its approach. We are unaware of any such documented rationale at present.

In making our assessments of the separate-effect test program, either completed or planned, it must be emphasized again that detailed descriptions of the test facilities and test programs were not available. This important factor must be considered when evaluating the conclusions presented in this section.

Assessment of Integral Test Data Base

The key PIUS systems, components, processes and phenomena were summarized in Table 6-2. One or more sufficiently prototypic integral tests are needed to develop the needed integral data base. Some of the testing can be accomplished in separate-effects tests that focus on specific systems, components, or key processes. This data base could then be integrated with the data bases from the integral testing program. Obviously, a plant prototype would provide the needed data base. ABB has stated that it believes a prototype plant is not needed (Ref. 7-1). Thus, ABB intends to satisfy the requirements of 10CFR52.47(b)(2)(i)(A) by completing appropriate test programs that (1) demonstrate the performance of each safety feature, (2) demonstrate interdependent safety effects among the safety features of the design, and (3) provide sufficient data on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal, transient, and accident conditions.

From Table 6-2, the key systems for which a data base is needed are (1) the density lock control, (2) plant protection based upon pool boron concentration, (3) the safety-related hydraulic behavior of the PIUS system, including the reactor internals, fuel assemblies, primary recirculation, pool systems, and scram-valve systems, and (4) the pool cooling system.

The ABB and Japanese proof-of-principle tests (Refs. 7-4 through 7-7) successfully demonstrated the density lock concept, i.e., that a stably stratified thermal layer could be created, maintained and controlled in a density lock component, thereby separating the primary system and a (potentially) heavily borated pool. The proof-of-principle tests were not designed with the intent of satisfying the requirements of 10CFR52.47(b)(2)(i)(A). They did not, for example, provide "sufficient data on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal, transient, and accident conditions." The focus of these tests was exploratory and the objective was proof of the PIUS density lock and natural-circulation cooling processes.

The ATLE test facility (Refs. 7-1 and 7-8) was constructed and operated in a manner that was more consistent with the requirements of 10CFR52.47(b)(2)(i)(A). Key features of the test are enumerated in the following.

1. The volume scale of ATLE is 1/308.
2. Key safety features were simulated.
3. Key components were simulated, including the upper and lower density locks, the reactor pool, pressurizer, core, riser, downcomer, reactor coolant pumps, and steam generators.
4. Key processes were simulated, including natural circulation through the upper and lower density locks, boron transport into the core (simulated with sodium sulfate), and control of the density locks interface. Core kinetics were *indirectly* simulated with a computer that calculated and controlled the core power based upon the solute concentration measured in the core region.

5. Key phenomena were simulated, including thermal stratification in the density locks.
6. The facility was instrumented and data recorded for code validation.
7. Tests completed in the ATLE facility included a load-following transient, a power-grid disturbance, a boron-front tracking test, an uncontrolled boron dilution event, a recirculation pump trip, and partial and total loss of heat sink. The data has been used to assess the ABB RIGEL code (Ref. 7-8) and the Los Alamos TRAC-PF1/MOD2 code (Ref. 7-9).

The ATLE facility and its operation had several important shortcomings and atypicalities. These are discussed briefly in the following.

1. The facility was scaled for the SECURE reactor and, therefore, differed from PIUS in several respects, including linear power, flow velocities, hydraulic arrangement, and height.
2. Several features of the PIUS design have evolved. The most important of these was the addition of the scram-valve system. This system was not present in the ATLE facility and in the tests previously conducted. Another important change was modifications to the density locks.
3. Additional atypicalities exist; these atypicalities have not been described. At full height and 1/308th volume scale, the facility is tall and narrow. In this regard, scaling compromises similar to those in the Semiscale (Ref. 7-10) and MIST (Ref. 7-11) facilities arise. ABB has not provided documentation describing the ATLE system and component modeling compromises, impact, and rationale to the NRC and Los Alamos. Issues related to the facility scaling and minimizing of scale distortions have been emphasized by the NRC in the AP600 review (see Section 2). A thorough and well-documented description of the scaling rationale, compromises, basis for the compromises, and impact is needed if data from the ATLE facility is to be used for validating the codes used for safety assessment in the certification stage.
4. Detailed drawings of the ATLE facility exist and were made available to the NRC and Los Alamos. The descriptions of the facility heat exchangers (representing the PIUS steam generators) were inadequate as the documented descriptions did not provide sufficient information to develop an accurate TRAC-PF1/MOD2 input model of this component. Schematics identifying the facility instrumentation and general location are available. The locations of the instrumentation was precisely defined. This information is important for finely noded models of the test facility so that comparisons of measured and calculated variables can be made near or at the same position. Descriptions of the ATLE facility operation for the tests are inadequate. For example, it is not clear from the documentation how the heat exchangers are controlled after a pump trip or partial loss of heat sink.
5. The most important physical processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain rodged shutdown and control systems. For safety-related reactor shutdown, PIUS relies on negative reactivity insertions from the moderator temperature coefficient (MTC) and boron. The boron enters the core after passing from the highly borated reactor pool and through the lower density lock. The ATLE test *directly* simulates neither of these reactivity processes that are closely coupled to PIUS thermal and hydraulic processes. ATLE simulates

these processes by processing moderator temperatures and solute concentrations through a point kinetics model. The resultant core power calculated in the point kinetics model is then provided as a core power setpoint to the power supply for the electrically heated core simulator rods.

6. ATLE does not simulate multidimensional effects. At the present time, it is not known whether such effects are important. In the review of the Simplified Boiling Water Reactor (SBWR), the NRC has expressed its concern that a component test facility did not represent the full-size system and important phenomena because of the one-dimensional nature of the component hardware tested (see Section 2). An important area of concern is related to item 5, above. The reactor core is not one dimensional; it has width and circumference. Should boron enter the core from the reactor pool in a nonuniform manner, the neutronic behavior of the core is uncertain. Undesirable phenomena, power-flow oscillations for example, could arise. The dimensional nature of the lower density lock, gas baffle region, natural-circulation inlet plenum, and core are not simulated.
7. The transients investigated in the facility to date, although important, do not cover a sufficient range of normal, transient, and accident conditions. Small-break and large-break loss-of-coolant accidents, loss-of-offsite power, and main steam line break transients were not simulated. ABB intends to simulate more severe transients in an ATLE supplementary test program (see Table 7-4).

ABB plans to conduct supplementary testing in a modified ATLE facility. The facility will be modified to better reflect the actual PIUS design in terms of hydraulic arrangement, water flow velocities, linear power, the scram-valve system, etc. (Ref. 7-3). A detailed description of the planned facility modification and test program is not available. A more extensive set of test transients will be tested, including large- and small-break loss-of-coolant accidents. Testing in this facility should address the shortcomings 1, 2, and 7 listed above. ABB will also be in a position to fully address shortcomings 3 and 4 by improving documentation of the facility scaling rationale and atypicalities, design, and operation. The modified ATLE facility will not address shortcomings 5 and 6, listed previously. In making these assessments, it must be emphasized again that a detailed description of the planned facility modification and test program is not available. This important factor must be considered when evaluating the conclusions presented in this section.

ABB has stated that it plans to construct a large-scale integral facility. The facility is to be full height and approximately 1/5 scale in diameter; this facility is also described as being about 1/15th in flow (Ref. 7-2, 7-3). A detailed description of the planned facility modification and test program is not available. The objective of the facility is to characterize the operational behavior of the complete PIUS system including density locks, reactor internals, reactor pool, wet thermal insulation, siphon breaker, pressurizer, and reactor coolant loops. If the facility is approximately 1/5th linear (1/15th flow) scale and accurately models the reactor internals, core, and primary recirculation system, it will potentially address the multidimensional thermal-hydraulic features of the PIUS reactor. The test program will include verification of start-up procedures from cold conditions to full temperature, normal operation including load following and operational transients, and shut-down procedures to hot and cold shutdown conditions. There is no mention of including transient and accident conditions in the test program. A large-scale facility that did not have the capacity to simulate transient and accident conditions would be of considerably diminished value. Evidently, ABB is considering several alternatives for further verification of the PIUS system behavior; one such alternative is "an intermediate size rig." (Ref. 7-3). In making these assessments, it must be emphasized again that a

detailed description of the planned facility modification and test program is not available. This important factor must be considered when evaluating the conclusions presented in this section.

ABB has stated that it plans an integral test of the reactor pool passive decay heat removal system (Ref. 7-2, 7-3). A detailed description of the planned facility modification and test program is not available. The planned test facility and test program will characterize the performance of the passive decay heat removal system including the water cooler immersed in the reactor pool, the intermediate closed cooling natural-circulation system, and the natural-draft air coolers. The planned facility will be full scale in elevation but the testing program will be conducted at reduced power. The test program will be performed at ambient pressure and will cover the full temperature range during normal operation, transients, and accidents. In making these assessments of the passive decay heat removal test program, it must be emphasized again that a detailed description of the planned facility modification and test program is not available. This important factor must be considered when evaluating the conclusions presented in this section.

If properly designed and conducted, the planned ABB test program has the potential, with one exception, of fulfilling the requirements to satisfy the requirements of 10CFR52.47(b)(2)(i)(A) by completing appropriate test programs that (1) demonstrate the performance of each safety feature, (2) demonstrate interdependent safety effects among the safety features of the design, and (3) provide sufficient data on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal, transient, and accident conditions. However, the one exception is of major importance. As previously indicated, the most important and most safety-related processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain rodged shutdown and control systems. For safety-related reactor shutdown, PIUS relies on negative reactivity insertions from the MTC and from boron. The boron enters the core after passing from the highly-borated reactor pool and through the lower density lock. None of the ABB-planned integral tests *directly* simulate either of these reactivity processes that are closely coupled to PIUS thermal and hydraulic processes.

With the presently available descriptions of planned integral testing programs, this remains an area of uncertainty. Tests conducted in the ATLE facility to date have provided valuable insights and data for code validation, even though the core neutronics and system thermal-hydraulic behaviors have been indirectly simulated through a computer model. It appears that the ATLE simulator approach could be extended to a multidimensional, electrically heated core simulator with individually controlled radial and azimuthal core regions. However, if such a test is envisioned, and only this type of test is envisioned to demonstrate PIUS coupled core neutronics and thermal-hydraulic behavior, it will be necessary for ABB to develop a convincing argument that this simulator approach provides sufficiently prototypical data. As discussed in Section 2, the NRC is concerned about the performance of passive safety systems that do not have a proven performance history. Given the design issues and performance uncertainties associated with passive safety features that do not have a proven performance history, testing is recognized by the NRC staff as the fundamental vehicle for addressing and reducing uncertainties to an acceptable level. No such testing has been identified that directly addresses the coupling of the reactor shutdown mechanisms and the core thermal and hydraulic processes.

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Table 7-1

PIUS Separate Effect and Key Process Tests Completed

Separate Effect and Key Process Tests	Test Description	Comments
Boron transport across thermal gradients Ref. 7-1, p. 14-6	Density lock high temperature Test created an arrangement where hot primary system water was stably layered above cold water.	Measured solute transport across the temperature gradient typical in a single density lock tube under prototypic pressure and temperature conditions. The facility was subjected to sloshing boundary conditions measured in a full scale tests. Tests confirmed boron transport can be kept acceptably low from an operational standpoint.
Stability of thermally stratified layers Ref. 7-1, p. 14-6	Small-scale experiments to demonstrate stability of thermal interface in a density lock. Full scale model of hot/cold "sloshing" in the density lock pipes exposed to representative turbulence conditions.	
PCRV Ref. 7-1, p. 14-3	Scandinavian PCRV Development Program 1967-1976.	ABB Atom states that feasibility has been demonstrated. PCRVs have been used in the British gas cooled reactors. A PCRV was used in the 330 MWe Fort St. Vrain reactor.
Wet thermal insulation Ref. 7-1, p. 14-5	Conducted in the PCRV test vessel described above. Insulation consisted of stainless steel mesh enclosed in layers of thin stainless steel sheets.	

Table 7-2

PIUS Separate Effect and Key Process Tests Planned

Test	Test Description	Comments
Stability of thermally stratified layers Ref. 7-1, p. 14-6	Full scale model of hot/cold "sloshing" in the density lock pipes exposed to representative turbulence conditions.	
Flow distribution in riser and upper density lock Ref. 7-2 Ref. 7-3	Test of the flow distribution in the riser at the connection to the upper density lock. Measure the detailed flow conditions in the riser and characterize the disturbance to the density lock from the reactor coolant flow.	Reduced scale test of upper density lock and relevant portion of riser planned for full coolant flow range at atmospheric pressure and temperature up to nearly boiling.
Density lock test Ref. 7-1, p. 14-9 Ref. 7-3	The tests will (a) characterize the transport mechanisms in the density lock, (b) verify the design features provided to reduce excessive mixing of primary coolant and pool water in the density lock, and (c) characterize the dynamic behavior of the density lock.	A full scale density lock module will be tested at ambient pressures. The influences of disturbances arising in both the primary and the pool and the pool will be simulated. The range of disturbances will span those anticipated for both the upper and lower density locks.
DNB test for PIUS fuel assembly Ref. 7-3	Verify the DNB correlation used in the analysis of normal operation, transients, and accidents.	
Transient flow conditions in the reactor pool Ref. 7-2 Ref. 7-3	Characterize transient flow conditions in the reactor pool following release of hot water through the upper density lock after trip of one or more of the reactor coolant pumps.	Approximately 1/10 scale test of mixing of hot and cold water in the pool at atmospheric pressures planned. Data will be used to validate the theoretical model used to calculate the mixing of the hot reactor water into the cold water in the pool. Data will also define performance requirements for the wet thermal insulation.

Table 7-2 (continued)

PIUS Separate Effect and Key Process Tests Planned

Test	Test Description	Comments
Siphon breaker Ref. 7-1, p. 14-10 Ref. 7-2 Ref. 7-3	Verify the hydraulic performance of the siphon breaker during normal operating conditions, including startup and shutdown. Verify the siphon breaking capability following primary system, cold-leg pipe breaks.	Full scale verification of detailed design at atmospheric pressures.. Optimize the detailed design. Sensitivity to pressure and temperature will be explored. Flow behavior will be examined. The hydraulic scaling laws will be studied in order to qualify the prototypicality of the siphon breaker modeling in scaled integral tests.
PCRv Ref. 7-3	Listed below are potential tests. (a) Test one or more full-circle tendons to demonstrate stress distribution. (b) Mechanical rebar splices (c) Test the venting ducts that could function as a crack pressure-relief system (e) Liner fabrication (f) Concrete compositions	ABB will consider the need for further tests related to the construction of the PCRv. ABB will also consider the need for an integral test of a model PCRv for validation of computer codes used in design and safety analyses.
Wet thermal insulation Ref. 7-1, p. 14-9 Ref. 7-3	Characterize the heat resistance of the insulation and heat losses in joints.	Tests will be performed with full scale panels of the wet insulation at nominal operating conditions as well as transients, including depressurization.
Reactor coolant pump Ref. 7-1, p. 14-12	Verification testing of one pump will be performed. Source: PSID, p. 14-12	Need to characterize pump coastdown characteristics. Need to characterize pump operational characteristics.
Steam generator Ref. 7-1, p. 14-10	Tests planned in collaboration with manufacture Source:PSID, p. 14-10.	A full-length tube model will be used to test thermal performance and part load operation.

Table 7-3

PIUS Integral Tests Conducted

Integral Tests	Test Description	Comments
<p>Proof of principle</p> <p>Ref. 7-4</p>	<p>ABB Atom has already conducted a proof of principle test in a facility named MAGNE</p>	<p>Integral test of key concepts, limited scaling rationale, limited instrumentation, may be missing other key features. Although many of the processes and phenomena unique to the concept are included, many others may not be included. The data from such tests may be used to demonstrate code tools can simulate the processes and phenomena but they lack sufficient prototypicality for benchmarking codes for design or safety analysis uses.</p>
<p>Proof of principle</p> <p>Ref. 7-5</p> <p>Ref. 7-6</p> <p>Ref. 7-7</p>	<p>Tests of an automatic pump speed control were conducted in a small (elevation difference between thermal centers of the lower and upper density locks about 3.5 m), atmospheric pressure test loop of the PIUS type. The maximum power of the electrically heated core is 28 kW. The stated objective of the test apparatus was to understand qualitatively the basic thermal-hydraulic phenomena in a PIUS-type reactor during normal operations and accident conditions. The facility included models of the lower and upper density lock, a heated core, riser, and pool. Heat was rejected to the secondary side of a heat exchanger. Startup and power ramping tests were conducted.</p>	<p>Ref. 7-4 described testing of a pump speed control based upon fluid temperature at the center of the lower density lock, the method used by ABB. A previous paper (Ref. 7-5) used a control based upon measuring differential pressure in the lower density lock. This method was less successful.</p> <p>The method of core power control was not clearly explained but appeared to be related to the flow from the "poison tank," which simulated the heavily borated reactor pool in the ABB PIUS design.</p> <p>Authors concluded that the safety characteristics of the PIUS-type reactor has been confirmed in a loss-of-feedwater simulation test.</p>

Table 7-3 (continued)

PIUS Integral Tests Conducted

Integral Tests	Test Description	Comments
<p>Small-scale integral test</p> <p>Ref. 7-1, p. 14-7</p> <p>Ref. 7-8</p>	<p>ABB Atom has conducted a small-scale, full-height integral test simulating the SECURE reactor in the ATLE facility. The volume scale of ATLE is 1:308. Key components simulated were the upper and lower density locks, the reactor pool, pressurizer, core, riser, downcomer, reactor coolant pumps, and steam generators. Key processes simulated included natural circulation through the upper and lower density locks, boron transport into the core (simulated with sodium sulfate), and control of the density locks interface. Core kinetics were simulated with a computer that then controlled the core power level. Key phenomena simulated included thermal stratification in the density locks.</p>	<p>Scaled facility, simulates many key components, processes, and phenomena. However, other key components, processes, or phenomena may not be simulated. With the small scale there may be significant facility atypicalities, e.g., lack of dimensionality. Such tests may be used to provide a partial benchmarking of codes for design or safety analysis applications. They are usually not, in themselves, sufficient to develop the data needed for code validation. The facility instrumentation was sparse.</p>

Table 7-4

PIUS Integral Tests Planned

Integral Tests	Test Description	Comments
Passive decay heat removal Ref. 7-2 Ref. 7-3	Characterize the performance of the passive decay heat removal system including the water cooler immersed in the reactor pool, the intermediate closed cooling natural-circulation system, and the natural-draught air coolers.	The test will be performed at ambient pressure and will cover the full temperature range during normal operation, transients, and accidents. Facility will be full scale in elevation; it will be tested at a reduced scale in power.
ATLE supplementary tests Ref. 7-3	Compliment and extend the test program already completed in the ATLE test facility (Table 7-3). Include testing of loss-of-coolant accidents.	Modify the ATLE facility to better reflect the actual PIUS design in terms of hydraulic arrangement, water flow velocities, and linear power.
Large-scale integral test Ref. 7-2 Ref. 7-3	<p>ABB Atom is planning to conduct a large-scale integral test program. The facility is described as full height and scaled at 1/5th diameter.</p> <p>The objective of the facility is to characterize the operational behavior of the complete PIUS system including density locks, reactor internals, reactor pool, wet thermal insulation, siphon breaker, pressurizer, reactor coolant loops, etc.</p> <p>The test program will include verification of startup procedures from cold conditions to full temperature, normal operation including load following and operational transients, and shutdown procedures to hot and cold shutdown conditions.</p>	<p>The large-scale system test is envisioned by ABB as a design verification test. Although the tests have not yet been planned, the facility design and operation would reflect the final design. The tests would be conducted before start of construction of the first plant.</p> <p>Scaled facility simulates most key components, processes, and phenomena at a scale that minimizes many facility atypicalities. Such tests may be used to provide the basis for validation of codes for design or safety analysis applications. By comparison with the processes and phenomena that occurred in the small-scale integral tests, the characterization of importance can be completed.</p>

Section 8

Assessment of Confirmatory Research Needs

As the NRC has engaged in precertification and certification reviews of the ALWRs, it has documented a rationale for the conduct of confirmatory research. The elements of this rationale were identified and summarized in Section 2. In the following paragraphs, each of eight elements in the rationale is described briefly and the applicability of this rationale to PIUS is assessed. This section is concluded with a summary assessment.

Rationale 1

The development of independent tests and analyses are needed to ensure that the safety-related characteristics of the design are identified and understood. Confirmatory research permits the NRC to develop an independent data base for assessing the safety performance of the advanced passive reactors and auditing the applicant submittals. *Assessment.* Independent tests and analyses are needed to ensure that the safety-related characteristics of the PIUS design are identified and understood. Confirmatory PIUS research would permit the NRC to develop an independent data base for assessing the safety performance of the PIUS reactor and the applicant's submittal.

Rationale 2

It is important to have a data base diverse from that of the applicant; including data arising from different scaling rationales and at different scales. PIUS is a new reactor and the existing data base is small. *Assessment.* A broadened and diverse data base is desirable for the PIUS reactor. This additional data base would be particularly valuable if it were obtained in test facilities of different scale than those generated by ABB for PIUS and, possible, in facilities designed to a different scaling rationale.

Rationale 3

The NRC has developed its own computer codes, which are used for independently assessing and auditing plant performance. Validation of the NRC's codes for advanced reactors is much less advanced than for the current generation of LWRs. Testing is needed to support the development and application of the NRC's analysis codes that are used to audit the applicant's submittals. *Assessment.* Validation of the NRC's codes for the PIUS reactor is much less advanced than for the current generation of LWRs. Validation of the NRC's computer codes for the PIUS application is necessary if the PIUS design is submitted for certification.

Rationale 4

Direct involvement in the planning and conduct of experimental programs and development and application of audit codes increases the knowledge of the NRC staff and enhances the quality and accuracy of the licensing review. Confirmatory testing and audit analyses permit the staff to develop insights and understanding of the unique features of the ALWRs. *Assessment.* Direct involvement by the NRC staff in the planning of a PIUS confirmatory testing program will assuredly increase the knowledge of the NRC staff and enhance the quality and accuracy of its licensing review of the PIUS reactor.

Rationale 5

There is an element of uncertainty with new designs, particularly with designs that depart in significant ways from the experience base accumulated through the years. Confirmatory research assists the NRC staff in verification of the margins used by the staff in its regulatory positions and acceptance criteria for the design. It is highly desirable that these margins be assessed. *Assessment.* The PIUS design departs in significant ways from the experience base previously accumulated. This is most evident with respect to the reactivity control systems but the density lock components and pool natural-circulation cooling are also noteworthy. Confirmatory research will assist the NRC staff to assess the margins used by the staff in its regulatory positions and acceptance criteria for the PIUS reactor.

Rationale 6

Confirmatory research can support the identification of unpredicted systems interactions. Confirmatory testing provides the NRC with the capability for exploring a range of conditions and resolving questions about unpredicted systems interactions. *Assessment.* Confirmatory testing of the PIUS reactor would provide a capability for exploring a range of conditions and resolving questions about unpredicted systems interactions.

Rationale 7

It is the responsibility of the applicant to provide appropriate systems data to demonstrate safety systems performance for transients up to the plant design basis. Confirmatory research can investigate systems behavior and phenomenological issues over a range of conditions up to and beyond the design basis. *Assessment.* There are a number of beyond design basis conditions that should be explored for the PIUS reactor. These include density lock blockages and pool boron dilution events. Such tests could be conducted by the NRC in a confirmatory integral facility and would be valuable in assessing both margins and systems interactions.

Rationale 8

Public confidence is enhanced when confirmatory research is conducted by an organization that is independent of the applicant. *Assessment.* A confirmatory research program for the PIUS reactor would play an important role in increasing public confidence regarding the safety of the PIUS reactor.

Summary Assessment

The NRC has developed a comprehensive rationale to support confirmatory research programs it has implemented for ALWRs currently undergoing certification reviews. There are eight elements in this rationale. Each of these elements has been reviewed and assessed for applicability to the PIUS reactor. Each of these elements has been assessed as applicable to the PIUS reactor. Therefore, confirmatory research for the PIUS reactor is directly and strongly supported by the rationale previously developed and applied by the NRC for other ALWRs.

Section 9

Summary

The PIUS advanced reactor is a 640-MWe PWR based on a concept developed by ABB. The PIUS reactor design is characterized by ABB as a concept, not a final design, because it is currently evolving in significant details. ABB submitted a PSID to describe the PIUS design and safety characteristics in 1990. ABB submitted a PSID Supplement to the NRC describing a number of important design changes in 1993.

A primary ABB design objective was to make the possibility of a core degradation accident extremely small. The PIUS reactor utilizes simplified, inherent, passive, or other innovative means to accomplish safety functions. Reactivity is controlled by coolant boron concentration and the temperature of the reactor coolant. There are no mechanical control rods. The core is submerged in a large pool of highly borated water and the core is in continuous communication with the pool water through pipe openings called density locks. There are no mechanical devices (e.g., valves) that separate the primary system and the pool. The primary coolant pumps are operated so that there is a hydraulic balance in the density locks (openings) between the primary coolant loop and the pool, keeping the pool water and primary coolant separated during normal operation. An active system, the scram-valve system, is provided as the primary scram system but the inherent scram system is always available.

Because the PIUS reactor utilizes simplified, inherent, passive, or other innovative means to accomplish safety functions, it is subject to the requirements of 10CFR52.47(b)(2)(i)(A). This regulation requires that the applicant adequately demonstrate the performance of each safety feature, interdependent effects among the safety features, and a sufficient data base on the safety features of the design to assess the analytical tools used for safety analysis. The applicant is permitted to support this demonstration by either analysis, appropriate test programs, experience, or an appropriate combination thereof.

The NRC has requested that Los Alamos National Laboratory assess the quality and completeness of the existing and planned data bases used by ABB to validate its safety analysis codes. The NRC also requested that Los Alamos assess other relevant data bases and evaluate the need for confirmatory integral systems testing by the NRC for the PIUS design.

An important aspect of the assessment has been an effort to identify and define standards against which adequacy of the existing and planned testing programs can be evaluated. Two perspectives were developed. The basis for the first perspective was documented in Section 2. Insights about data base requirements and needs were identified based upon a licensing certification perspective. In ALWRs, of which PIUS is one concept, the well understood active safety-class systems of current commercial reactors are replaced with passive systems that do not have a proven performance history. The NRC is currently reviewing the AP600 and SBWR advanced passive reactors for design certification. Documentation from these reviews was examined to ascertain the level and type of testing that the NRC believes is necessary to support the certification of advanced passive reactors.

The basis for the second perspective was documented in Section 3. Perspectives about data base requirements and needs were identified by considering the need for validated computational tools (code) to predict the behavior of the advanced light water reactors during representative operational and accident sequences. This framework was developed

within the context of the CSAU effort pioneered by the NRC for the current commercial reactors. Within this framework, the reactor is identified (PIUS) and the key safety-related features of PIUS are also identified (see Section 4; key operational and accident scenarios are identified (see Section 5); and the key systems, components, processes, and phenomena associated with PIUS and the representative scenarios are identified (see Section 6).

Finally, the existing and planned data bases, whether developed by ABB or elsewhere, were reviewed for adequacy using the two perspectives described above. Los Alamos has concluded that only a limited data base of separate effect and integral tests exist at present. This data base is not adequate to fulfill the requirements of 10CFR52.47(b)(2)(i)(A). ABB has stated that it plans to conduct more separate effect and integral test programs. If completed, these test programs would satisfy most of the data base requirements of 10CFR52.47(b)(2)(i)(A) and remedy most of the deficiencies of the currently existing combined data base.

However, the most important physical processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain mechanical shutdown and control systems. For safety-related reactor shutdown, PIUS relies on negative reactivity insertions from the moderator temperature coefficient and from boron entering the core from the reactor pool. ABB has neither developed an experimental data base for these important processes nor provided a rationale for indirect testing of these key PIUS processes. This is assessed as a significant shortcoming.

In preparing the conclusions of this report, test documentation and results have been reviewed for only one integral test program, the small-scale integral tests conducted in the ATLE facility. Test design and data documentation have not been reviewed for the remainder of the separate-effect and integral test programs that have been completed to date, because Los Alamos did not request that ABB provide this information. Only brief outlines describing future programs have been provided and reviewed. The paucity of information about the ABB testing programs is an important factor that must be considered when evaluating the conclusions presented in this document.

As the NRC has engaged in precertification and certification reviews of the ALWRs, it has documented a rationale for the conduct of confirmatory research. The elements of this rationale were identified and summarized in Section 2. The elements in the rationale are as follows. First, the development of independent tests and analyses are needed to ensure the safety-related characteristics of the design are identified and understood. Second, it is important to have a diverse data base from that of the applicant; including data arising from different scaling rationales and at different scales. Third, testing is needed to support the development and application of the NRC's analysis codes that are used to audit the applicant's submittals. Fourth, direct involvement in the planning and conduct of experimental programs and development and application of audit codes increases the knowledge of the NRC staff, and enhances the quality and accuracy of the licensing review. Fifth, confirmatory research assists the NRC staff in verification of the margins used by the staff in its regulatory positions and acceptance criteria for the design. Sixth, confirmatory research can support the identification of unpredicted systems interactions. Seventh, confirmatory research can investigate systems behavior and phenomenological issues over a range of conditions up to and beyond the design basis. Eighth, public confidence is enhanced when confirmatory research is conducted by an organization that is independent of the applicant. The need for confirmatory integral systems testing by the NRC for the PIUS design has been assessed against each of the eight elements previously summarized in this paragraph. The author concludes that the conduct of confirmatory research for the PIUS reactor is supported by each of the eight rationale elements.

Therefore, confirmatory research for the PIUS reactor is directly and strongly supported by the rationale previously developed and applied by the NRC for other ALWRs.

Appendix A

Code Requirements

In this Appendix, requirements for thermal-hydraulic system analysis codes are identified and summarized. The specific intent of this Appendix is to identify requirements that will, if satisfied, provide computational tools capable of fulfilling the NRC's safety analysis needs in the area of PIUS system thermal hydraulics. The detailed justification and appropriate references for each requirement identified can be found in Ref. A-1. Some additional effort may be required to update the requirements to reflect current NRC policies and practices.

A.1. General Design Features

This listing of general design features is not intended to be all inclusive. Rather, it is intended to focus on several key code features that will have a major impact on the determination of code adequacy for the selected applications.

A.1.1. Best-Estimate Capability

The system code should be a best-estimate tool and should use modeling that attempts to realistically describe the physical processes occurring in the reactor. A best-estimate code should contain all the models necessary to predict the important phenomena occurring in the events that must be analyzed to ensure safety. After 15 years of thermal-hydraulic research in support of its LWR regulatory and safety activities, the US Nuclear Regulatory Commission concluded: "Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations." There is a companion requirement that the uncertainty in the realistic evaluation model be quantified and considered when comparing the results of the calculations with the applicable limits so that there is a high probability that the criteria will not be exceeded. The reliance on best-estimate codes to simulate thermal-hydraulic behavior of a nuclear facility is predicated upon three factors. First, it is too costly and unsafe to test the thermal-hydraulic response of DBEs using operating facilities. Second, one cannot directly apply results from test facilities because of scaling compromises in their design. Third, the study of various plant recovery techniques can only be performed by computer codes.

A.1.2. Thermal-Hydraulic Capabilities

The primary usage of a thermal-hydraulic system analysis code is to develop an understanding of and confidence in predictions of plant thermal-hydraulic performance during AOOs and DBEs. The computer codes are the principal products applied in an effort to understand and predict plant response to deviations from normal operating conditions. For AOOs and DBEs, a specific set of requirements can be developed. In addition, thermal-hydraulic systems codes have been used to provide the front end predictions of both deterministic and probabilistic analyses and for accident management studies. For such analyses a broader set of requirements may be appropriate. For example, it is possible that PIUS design criteria will preclude voiding in the primary system (with the exception of the pressurizer), even for DBEs. Nevertheless, the capability to analyze two-phase flow situations may still be required for certain BDBEs with intact core or for the period of the accident while the core remains intact. Los Alamos believes that the incorporation of such requirements within the specification of requirements for the PIUS thermal-hydraulic systems code is warranted. However, the requirements presented herein

are restricted to PIUS system thermal-hydraulic behavior up to, but not including, the point where fuel damage would begin to occur, even in support of severe accident studies.

A.1.2.1. Reactor Thermal-Hydraulic Design. The design bases for the thermal and hydraulic design of the PIUS will be reviewed including the requirements to calculate such items as maximum fuel and clad temperatures as a function of burnup (at rated power, at design overpower; and during transients); critical heat flux ratio (at rated power, at design overpower, and during transients), flow velocities, flow distribution, coolant and moderator voids; hydraulic stability; transient limits; fuel cladding integrity criteria; and fuel assembly integrity criteria.

A.1.2.2. Normal and Postulated Events. The evaluation of the safety of PIUS will include review and analyses of the response of the reactor to a broad set of initiating events arising during normal operation and from postulated disturbances in process variables and malfunctions or failures of equipment. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, the establishment of safety system settings, and the establishment of design specifications for components and systems important to public health and safety.

A.1.2.3. Beyond Design Basis Events with Intact Core. Evaluation of the safety of PIUS will also include review and analyses of initiating events and event sequences that, without careful analysis, would normally be assumed to lead to severe core damage. These BDBEs arise from two sources: initiating events of lower frequency than those that are required to be considered as DBEs or, for more likely initiators, event sequences that contain combinations of faults beyond the design basis fault. The BDBEs can be divided into two classes: severe BDBEs and nonsevere BDBEs. Severe BDBEs culminate in severe core damage, a condition that results in loss of a definable or coolable core geometry or a release of radioactive material from the core that, if uncontained, would constitute a large release. Nonsevere BDBEs can experience core damage but the damage does not result in a loss of a definable or coolable core geometry. The analysis requirements for nonsevere BDBEs can be more extensive than for AOOs and DBEs. For example, extensive fuel assembly voiding may occur in nonsevere BDBEs while similar phenomena are precluded, by design specification requirements, for AOOs and DBEs. Frequently, however, the same analysis tools can be used for AOOs and DBEs and for nonsevere BDBEs. Therefore, the capability to analyze nonsevere BDBEs is included as part of the thermal-hydraulic systems analysis code requirements. The phenomena associated with severe BDBEs do not contribute to the requirements for thermal-hydraulic systems analysis codes covered by this document.

A.1.3. Adequate Numeric Performance

Systems codes have three key elements; they are the architecture, modeling of physical processes, and numerical methods. If the numerical methods are adequate, they ensure the accurate solution of the governing equations (modeling of physical processes) within acceptable bounds. The numerical methods used must be tested and documented.

A.1.4. Total Analysis Costs

The total analysis costs are the summation of four components (1) the cost to develop the plant model/input file, (2) the cost associated with getting to the final calculation (e.g., how difficult is it to get the model right and produce the calculation that is retained as the answer), (3) the cost per final calculation, and (4) the cost to analyze and interpret the results. If the total analysis costs are too great to support financially, the other qualities of a computer code become immaterial. The concept of value should also be considered. As

used here, value is a combination of the quality of the technical result produced and the cost required to produce that result.

A.2. Modeling Capabilities

The requirements listed in the preceding section described several general design features; these focused on the fundamental modeling philosophy (best-estimate), thermal-hydraulic event sequence capability (reactor thermal-hydraulic design, anticipated and design basis events, and beyond design basis events with intact core), adequate numeric performance, and total analysis costs. The current section focuses on the modeling features to be incorporated in the thermal-hydraulic systems analysis code for Los Alamos safety verification analyses. The modeling capabilities requirements are grouped according to the following structure.

- Component Modeling
- Systems Modeling
- State Modeling (Steady and Transient)
- Fluid and Material Properties
- Modeling of Key Phenomena

A.2.1. Components

The code must have the capability of modeling PIUS. In addition, the code must be validated using experimental data from separate effect and integral test facilities. To satisfy these two needs the code must be able to model the elements of both the reactor and experimental facilities and combine these into an integrated plant or facility model. It is required that the code have the capability to model fuel assemblies, the core, plena, reactor coolant pumps, heat exchangers, vessels, pressurizers, pipes, valves, heat structures, density locks, and the siphon breaker. Specific requirements for each of these components follow.

A.2.1.1. Fuel Assemblies. The fuel assemblies are the first barrier to fission product release in a defense in depth philosophy. The code must have the capability to model the coupled fluid dynamic and heat transfer processes as they apply to the geometry, composition, and power generation features of the fuel assemblies. The code must have the capability to model the dominant processes and phenomena in the fuel assemblies. The code must have the capability to calculate the thermal-hydraulic parameters associated with the appropriate detailed acceptance criteria specified for the fuel assemblies. The code must also have the capability to model fuel assembly simulators used in test programs.

A.2.1.2. Core. The code must have the capability to model not only the thermal-hydraulic performance of the fuel assemblies, but also the performance of the entire core. The code must have the capability to model the dominant processes and phenomena in the core.

A.2.1.3. Plena. The code must have the capability to model the dominant processes and phenomena in the plena should they occur in the design.

A.2.1.4. Reactor Coolant Pumps. Reactor coolant pumps will be present in PIUS, and the code must have the capability to model the dominant processes and phenomena in these pumps. The code must be able to accurately simulate the behavior of these pumps in normal and off-normal operating regimes. Pump coastdown, operation during two-phase conditions with noncondensable gases, and cavitation must be modeled.

A.2.1.5. Steam Generators/Heat Exchangers. Steam generators and heat exchangers (pool coolers) will be present in PIUS and the code must have the capability to model the dominant processes and phenomena in these components. The code must be able to accurately simulate the behavior of these components in normal and off-normal operating regimes. Operation during two-phase conditions with noncondensable gases must be modeled.

A.2.1.6. Vessel Including Riser and Downcomer. The vessel is part of the second barrier to fission product release in a defense-in-depth philosophy. The code must have the capability to model the coupled fluid dynamic and heat transfer processes as they apply to the vessel and primary system piping and the dominant processes and phenomena in the vessel. The code must also have the capability to model vessels used in test programs.

A.2.1.7. Pressurizer. A pressurizer will be present in PIUS and the code must have the capability to model the dominant processes and phenomena in this component. The code must be able to accurately simulate the behavior of this component in normal and off-normal operating regimes. Operation during two-phase conditions with noncondensable gases must be modeled.

A.2.1.8. Pipes. The primary system piping is part of the second barrier to fission product release in a defense-in-depth philosophy. The code must have the capability to model the coupled fluid dynamic and heat transfer processes as they apply to the system piping and the dominant processes and phenomena in pipes. The code must also have the capability to model vessels, system piping, and valves used in test programs.

A.2.1.9. Valves. Some valves, e.g., relief valves, are present in the PIUS design. The code must have the capability to model the dominant processes and phenomena in these components. The code must also have the capability to model valves used in test programs.

A.2.1.10. Heat Structures. There are circumstances where the heat transfer from structural materials in the reactor can materially affect the transient path. The code must include the capability to model the dominant heat-transfer processes and phenomena associated with key structures.

A.2.1.11. Upper Density Lock. The upper density lock is a component unique to the PIUS design. In addition, it is a key component in that it is an integral part of the PIUS reactivity control philosophy. It is essential that the capability to model this component be provided.

A.2.1.12. Lower Density Lock. The lower density lock is a component unique to the PIUS design. In addition, it is a key component in that it is an integral part of the PIUS reactivity control philosophy. It is essential that the capability to model this component be provided.

A.2.1.13. Siphon Breaker. The siphon breaker is included in the PIUS design to ensure that sufficient coolant remains in the core to cool the core following loss-of-coolant accidents. Thus, the siphon breaker is a key component for ensuring core cooling. It is essential that the capability to model this component be provided.

A.2.2. Systems

A systems code must be able to successfully calculate the behavior of the total system. In addition to the physical systems described in Section A.2.1, there are plant systems such as the control and protection systems that must be modeled. Ultimately, the total code must successfully integrate the models of the PIUS components, control systems, and plant protection systems.

A.2.2.1. Control Systems. Nuclear reactors employ control systems to set limits on the course of transient behavior during normal events. For some events requiring analyses, these control systems operate during all or part of the transient. Therefore, the systems code must have the capability to model the control system logic.

A.2.2.2. Plant Protection Systems. Nuclear reactors employ plant protection systems to set limits on the course of transient behavior during off-normal events. Therefore, the systems code must have the capability to model the plant protection system trip and control logic.

A.2.2.3. Systems Integration. Requirements for modeling individual PIUS components were identified in the preceding sections. Because the objective of the code is to provide an integrated system transient analysis capability, the code must effectively couple the calculated component behavior, control and plant protection systems, and operator actions in a physically realistic manner.

A.2.3. State Modeling

There are two states of interest—steady and transient. Of the two, the ability to model the transient state is of the greatest importance. However, since the largest fraction of transient analyses is initiated from steady-state operating conditions, the capability to calculate the steady state is also important.

A.2.3.1. Steady State. All transient analyses start from an initial condition, which is usually the steady-state operating condition of the plant. If the initial steady-state configuration for transient analyses cannot be produced by the transient analysis code, then it has to be developed otherwise and then input to the code. For these reasons it is prudent to require that the transient analysis code have the capability to self-generate the initial conditions for the transient calculation, which are usually steady-state conditions.

A.2.3.2. Transient. The primary objective of the code is to provide a tool for transient analysis. Thus, the requirement that the code be able to calculate time-varying thermal-hydraulic phenomena is fundamental.

A.2.4. Multi-Dimensional Modeling

In both the current LWRs and the Savannah River Reactors, the ability to simulate key multidimensional phenomena in the reactor vessel has been necessary. Until it can be determined if a multidimensional modeling capability is required for analysis of PIUS accidents, it is required that such a capability be provided for this code.

A.2.5. Reactor Kinetics Modeling

Core power is the key physical force driving the need for an accurate, predictive, transient and accident analysis tool. The core power is coupled to the thermal and hydraulic

processes occurring in the core. It is required that the code have the capability of modeling the core physics processes and phenomena. It will be necessary to determine if multidimensional modeling of the reactor kinetics is required.

A.2.6. Fluid and Material Properties

The code must have the capability of modeling the specific working fluids and materials in PIUS. In addition, the code must be validated using experimental data from separate effect and integral test facilities, which may be constructed of different materials and use different fluids. Specific requirements and justification for each of these materials follow.

A.2.6.1. Light Water—Single- and Two-Phase. The code shall have the capability to model the physical properties of single- and two-phase light water.

A.2.6.2. Noncondensable Gas. The code must have the capability to model noncondensable gas behavior including transport and influence on heat transfer.

A.2.6.3. Structural Materials. The code shall have the capability to model the physical properties of those structures whose presence will have a significant influence on the path of key transients.

A.2.7. Key Phenomena

The code must have the capability to model important accident sequences, including all limiting accident sequences, and the phenomena that occur during those sequences. Only a limited amount of effort has been expended in this area. However, this work has resulted in important insights. The most applicable work to date is summarized in Section 6 of this report. In addition to the comprehensive list provided in Table 6-1, an effort has been made to identify the key phenomena are these are listed Table 6-2. A discussion is provided in Section 6 that describes the rationale used in identifying key processes and phenomena.

A.3. Functional and User Interface

Requirements related to the functional and user interface are universally understood to be important. Functional and user interface requirements were recently documented as part of a DOE-sponsored effort to develop criteria for the selection of codes to be used in severe accident analysis for the Advanced Light Water Reactor (ALWR). These requirements constitute an excellent summary for not only severe accident code features and efforts but also the PIUS thermal-hydraulic systems codes. The three areas of functional and user interface highlighted are documentation, adaptability and maintainability, and ease of use and interpretation.

A.3.1. Documentation

The documentation of the codes shall be publicly available and shall clearly describe the codes and models that are included according to an appropriate standard (e.g., IEEE Standard 730-1984). This provides additional acceptance and understanding of the technical content of the code and additionally encourages the use of the package by others. This is an essential element of the certification process. Review and acceptance of the methodology by the sponsoring organization is a necessary objective of the program. The major criteria that will be used to measure how well the codes meet this objective are: (1) the present status of the validation and benchmarking, (2) the availability and content of

a user's guide, (3) the availability and content of a model description, and (4) the code verification report.

A.3.2. Adaptability and Maintainability

The code shall be easily maintained and shall be designed to provide practical configuration management. An archival version of the code must be available. The code will be written such that the modules are structured and form a logical hierarchy. In addition, the code will be written in modular fashion. This feature allows the developer to make changes in individual subroutines without affecting other parts of the program in a significant way. The code shall provide easy adoption and interface with models and databases for thermophysical and chemical properties.

A.3.3. Ease of Use and Interpretation

A major factor in the usefulness of an accident analysis computer code is the amount of effort required to apply it to various nuclear power plant and experimental configurations and to interpret the results of the analyses when completed. In order to achieve this objective, the definition of the problem being considered must be easily modified by the user. A flexible input capability can provide a means to accomplish this and allows the user a desirable degree of control over the problem being considered. The code must provide the user with complete output in a form that allows easy interpretation of results, including both tabular and graphical presentation of data.

A.4. Verification and Validation

Verification and validation activities should cover the following areas. First, the applicability, range, and scale-up capability of the models and correlations should be established. The models and correlations used should be traceable to their source. Second, verification of the coding of equations and models should be provided. Third, assessment of the code-calculated results against available separate effect test data and integral system test data should be carried out and documented. Comparisons to the results of calculations with diverse codes is also useful. Fourth, there should be an independent review and assessment of the models and correlations, component models, and plant models. Fifth, the uncertainty of the code calculation should be quantified.

A.4.1. Individual Models and Correlations

The adequacy of each model is built upon a defensible pedigree and applicability to a given PIUS application. However, adequacy has the additional feature that the representation or prediction of key phenomena is demonstrated to be acceptably good. This fidelity can be demonstrated if the model correctly predicts the major trends of the dominant PIUS phenomena being modeled, and the uncertainty can be quantified and is acceptably small. Thus, a key process for certifying the adequacy of a model is by validating the model using an appropriate set(s) of experimental data. A secondary method of certifying the adequacy of a model is to consider the completeness and adequacy of documented comparisons to other validated models.

A.4.2. Integrated Code

Not only must the individual models be validated, but the total code must be validated using appropriate data to ensure that all the separate elements of the code, including the individual models and correlations, have been successfully integrated. Thus, a key process for certifying the adequacy of a method is by validating the total code using an appropriate set(s)

of integral test experimental data. A secondary process of certifying the adequacy of a method is to consider the completeness and adequacy of documented comparisons to other validated codes.

A.5. Quality Assurance

Quality assurance must be an integral part of any code used by the NRC to perform safety analyses in support of its mission to assure the public health and safety.

References

- A-1. B. E. Boyack and J. S. Elson, "Assessment of TRAC-PF1/MOD3 Code Adequacy for NP-HWR Thermal-Hydraulic Analyses," Los Alamos National Laboratory Advanced Reactor Safety Group document LA-NPR-TN-010 (September 15, 1992).

Appendix B

Summary Tabulations

The NRC has been engaged in preapplication reviews of several reactors, the modular high temperature gas cooled reactor or MHTGR; a liquid metal reactor, the PRISM reactor; a heavy water moderated reactor, the CANDU 3 reactor; and the PIUS reactor. The NRC has sponsored evaluations of the quality and completeness of the experimental data bases for each reactor type and requested that the outcomes of each of these data base evaluations be compiled in a common format. In this Appendix, the information presented in the body of this report is reformatted and presented in the requested summary tables. Brief perspective statements are provided in the following paragraphs to assist the reader in evaluating the content of the summary tabulations. The table numbering system is that requested by the NRC with the exception that each table number is started with "B" to follow the numbering convention of this report and denote that the table belongs to Appendix B.

PIUS Representative Sequences

PIUS representative sequences for Event Category I (EC-I) are presented in Table B-3.2.1. EC-I events are considered to be Anticipated Operational Occurrences or AOOs. The starting list of representative EC-I events is provided in Table 5-1. The events in Table 5-1 are "representative" in that they are thought to include all the major phenomena expected to occur during AOO events. However, neither the listing of events in Table 5-1 nor the more abbreviated list of events in Table B-3.2.1 should be considered to be complete listings of all AOO events that may occur in the PIUS reactor. For example, three event sequences are included in Table 5-1 but are not included in Table B-3.2.1. These are (1) loss of a single reactor coolant pump, (2) feedwater transients, and (3) loss of reactor pool cooling. A loss-of-offsite power (LOSP) event results in the loss of all four reactor coolant pumps, and culminates in fully natural-circulation flows from the reactor pool, through the lower density lock, through the reactor core and riser, and a return to the pool through the upper density lock. The loss of a single reactor coolant pump displays a mixture of phenomena because the scram-valve system operates and a combination of forced and natural circulation exists. Feedwater transients can result in either undercooling or overcooling events. A turbine trip event is an undercooling event and, therefore, not covered separately in Table B-3.2.1. The loss of [active] reactor pool cooling is a benign event because the reactor pool has a large capacity for absorbing heat, and a completely passive cooling system will cool the pool following loss of the active pool cooling systems.

PIUS representative sequences for Event Category II (EC-II) are presented in Table B-3.2.2. EC-II events are considered to be DBEs. The starting list of representative EC-II events is provided in Table 5-1. The events in Table 5-1 are "representative" in that they are thought to include all the major phenomena expected to occur during DBE events. However, neither the listing of events in Table 5-1 nor the more abbreviated list of events in Table B-3.2.2 should be considered to be complete listings of all DBE events that may occur in the PIUS reactor. The reactor scram with failure of the scram valves to open is not presented because the event is bounded by events EC-I-1 and EC-I-2 from Table B-3.2.1.

Data Types Useful for Confirming Key Phenomena

Data types useful for confirming key phenomena in the areas of physics and thermal-hydraulics for representative PIUS sequences in EC-I are presented in Tables B-3.3.1.1 and B-3.3.1.2. Data types useful for confirming key phenomena in the areas of physics and thermal-hydraulics for representative PIUS sequences in EC-II are presented in Tables

B-3.3.2.1 and B-3.3.2.2. The parameter ranges of data are difficult to define at this time because the PIUS design is at the conceptual stage. However, important attributes of the needed data are addressed in the tabulations. One source of information regarding the parameter ranges and instrumentation needs could be transient analyses of each of these events performed with analysis codes such as RIGEL or TRAC-PF1/MOD2.

Integral Data Types Useful for Confirming Key Phenomena

Integral data types useful for confirming key phenomena in the areas of physics and thermal-hydraulics for representative PIUS sequences in EC-I are presented in Tables B-3.4.1.1 and B-3.4.1.2. Data types useful for confirming key phenomena in the areas of physics and thermal-hydraulics for representative PIUS sequences in EC-II are presented in Tables B-3.4.2.1 and B-3.4.2.2. The parameter ranges of data are difficult to define at this time because the PIUS design is at the conceptual stage. However, important attributes of the needed data are addressed in the tabulations. One source of information regarding the parameter ranges and instrumentation needs could be transient analyses of each of these events performed with analysis codes such as RIGEL or TRAC-PF1/MOD2.

Existing Data Base Information for the PIUS Reactor

The existing data base for the PIUS reactor is small. Separate effect and key process tests completed to the present are tabulated in Table 7-1 of this report. Integral tests completed to the present are tabulated in Table 7-3 of this report. Planned separate effect and key process tests are tabulated in Table 7-2 of this report. Planned integral tests are tabulated in Table 7-3 of this report. Descriptions of the planned separate effect and key process tests and integral tests are very brief and, therefore, these tests are characterized as "potentially adequate." This means that the tests have the potential for being adequate if they are appropriately designed and conducted. The references cited in Tables B-3.5.1.1, B-3.5.1.2, B-3.5.2.1 and B-3.5.2.2 are presented at the end of this section.

Prioritized Areas for PIUS Data Base Enhancements

Prioritized areas for PIUS data base enhancements are presented in Table B-3.6.1 for EC-I and EC-II events. Although there are many similarities for the two lists, there are also important differences. These are primarily related to the severity of the event and the existence of voiding in the primary system.

Prioritized Areas of Missing or Partial PIUS Data

See Tables B-3.6.1 and B-3.6.2. The author has been unable to distinguish between data bases needing enhancements and partial data bases.

References

- B-1. "PIUS Preliminary Safety Information Document," ABB Atom (December 1989).
- B-2. "PIUS Conceptual Design Review," Asea Brown Boveri/Combustion Engineering Handout, Meeting with NRC Staff (March 19, 1991).
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Table B-3.2.1

**PIUS Representative Sequences of Event Category I
(Anticipated Operational Occurrences)**

NUMBER	VENDOR ID	EVENT DESCRIPTION
EC-I-1		Reactor scram with scram-valve system
EC-I-2		Loss-of-offsite power (all reactor coolant pumps)
EC-I-3		Feedwater transients
EC-I-4		Primary loop boron dilution

Table B-3.2.2

**PIUS Representative Sequences of Event Category II
(Design Basis Events)**

NUMBER	VENDOR ID	EVENT DESCRIPTION
EC-II-1		Reactor scram with (a) pool boron concentration at 1800 ppm, or (b) partial blockage of the lower density lock
EC-II-2		Steam line break
EC-II-3		Small primary pipe break
EC-II-4		Large primary pipe break

Table B-3.3.1.1

Data Types Useful for Confirming Key Phenomena in the Area of Reactor Physics for Representative PIUS Sequences in Event Category-I

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR	REPRESENTATIVE EC-I SEQUENCES (See Table B.3.2.1)				COMMENTS
	CONFIRMING	1	2	3	4	
	PHENOMENA	PARAMETER RANGES OF DATA				
Core power	Reactivity vs moderator temperature	BOC, EOC and midrange	Worst case	Worst case	Worst case	Lacking mechanical control and shutdown rods, characterization of core neutronics and reactivity insertions is essential.
	Reactivity vs boron concentration	BOC, EOC and midrange	Worst case	Worst case	Worst case	
	Spatial core variations due to moderator temperature and boron concentration	BOC, EOC and midrange	Worst case	Worst case	Worst case	Spatial effects could be measured in a criticals experiment under quasi-static conditions

BOC = Beginning of cycle.

EOC = End of cycle.

Table B-3.3.1.2

Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-I

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-I SEQUENCES (See Table B.3.2.1)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Coolant flow (a) forced (b) natural/ buoyancy induced (c) transitions - pump coastdown - density lock activation (d) oscillatory	Pressure distribution	Primary at forced circulation	Primary at forced and natural circulation	Primary at forced circulation	Primary at forced circulation	<u>Components</u> (a) downcomer (b) core (c) riser (d) lower density lock
	Temperature distribution	Primary system temperatures	Primary system temperatures	Primary system temperatures secondary temperatures	Primary system temperatures	(e) upper density lock (f) reactor coolant pump (g) forced and natural circulation inlet plena
	Flow rate	Primary system flows	Primary system flows	Primary system flows Secondary flows	Primary system flows	<u>Systems</u> (a) primary (b) pool coolant (c) scram-valve
	Flow regime	Single phase	Single phase	Single phase	Single phase	

Table B-3.3.1.2 (continued)

Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-I

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-I SEQUENCES (See Table B.3.2.1)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Boron distribution and concentration	Concentration vs position in prototypic geometry and mixing streams	Primary system	Primary system	Primary system	Primary system	Distribution is a function of transport and mixing of the fluid streams. Concentration is a function of diffusion and mixing of fluid streams of differing concentrations
Thermal interface control	position, stability, recovery	Throughout transient	NA	Throughout transient	Throughout transient	Control is essential for separation of primary coolant and highly-borated pool

Table B-3.3.2.1

Data Types Useful for Confirming Key Phenomena in the Area of Reactor Physics for Representative PIUS Sequences in Event Category-II

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
	CONFIRMING	1	2	3	4	
	PHENOMENA	PARAMETER RANGES OF DATA				
Core power	Feactivity vs moderator temperature	BOC, EOC and mid-range	Worst case	Worst case	Worst case	Lacking mechanical control and shutdown rods, characterization of core neutronics and reactivity insertions is essential.
	Reactivity vs boron concentration	BOC, EOC and mid-range	Worst case	Worst case	Worst case	
	Reactivity vs void	NA	NA	Worst case	Worst case	Void does not occur in all events
	Spatial core variations due to moderator temperature, boron concentration, and void	BOC, EOC and mid-range	Worst case	Worst case	Worst case	Spatial effects could be measured in a criticals experiment under quasi-static conditions

BOC = Beginning of cycle.

EOC = End of cycle.

Table B-3.3.2.2

Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-II

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Coolant flow (a) forced (b) natural/ buoyancy induced (c) transitions - pump coastdown - core uncovery - core recovery - density lock activation - fluid phase (d) oscillatory	Pressure distribution	Primary at forced circulation	Primary at forced circulation	Primary at forced and natural circulation	Primary at forced and natural circulation	<u>Components</u> (a) downcomer (b) core (c) riser (d) lower density lock
	Temperature distribution	Primary system temperatures	Primary system temperatures Secondary temperatures	Primary system temperatures	Primary system temperatures	(e) upper density lock (f) reactor coolant pump (g) forced and natural circulation inlet plena
	Flow rate	Primary system flows	Primary system flows Break flow	Primary system flows Break flow	Primary system flows Break flow	<u>Systems</u> (a) primary (b) pool coolant (c) scram valve
	Flow regime	Single phase	Single phase	Single phase two phase vapor	Single phase two phase vapor	

Table B-3.3.2.2 (continued)

Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-II

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Boron distribution and concentration	Concentration vs position in prototypic geometry and mixing streams	Primary system	Primary system	Primary system	Primary system	Distribution is a function of transport and mixing of the fluid streams. Concentration is a function of diffusion and mixing of fluid streams of differing concentrations
Blockage	Flow vs blockage fraction	50-75% blockage of lower density lock	NA	NA	NA	Impacts coolant flow, boron distribution & concentration.
Siphoning	Flow termination	NA	NA	Worst SBLOCA	Worst LBLOCA	Siphon must be broken under LOCA conditions to prevent draining of pool
Thermal interface control	Position, stability, recovery	Throughout transient	Throughout transient	NA	NA	Control is essential for separation of primary coolant and highly-borated pool

Table B-3.4.1.1

Integral Data Types Useful for Confirming Key Phenomena in the Area of Physics for Representative PIUS Sequences in Event Category-I

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-I SEQUENCES (See Table B.3.2.1)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Core Power	Spatial core variations due to moderator temperature and boron concentration	BOC, EOC and mid-range	Worst case	Worst case	Worst case	The normal reactivity control systems should also be demonstrated under normal operating conditions Dynamic coupling, in contrast to quasi-static coupling, should be demonstrated Normal power peaking factors at transient initiations should be included

Table B-3.4.1.2

Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-I

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.1)				COMMENTS
	CONFIRMING	1	2	3	4	
	PHENOMENA	PARAMETER RANGES OF DATA				
Pumped Coolant Flows	Flow rates	Primary system flows	Primary system flows	Primary system flows	Primary system flows	Trips using the scram-valve system require forced circulation
	Flow paths	Pressure distributions	Pressure distributions	Pressure distributions	Pressure distributions	Accurately characterize primary flow resistances and pump performance
Natural Circulation Coolant Flow	Flow rates	NA	Primary system flows	NA	NA	Make provisions for resolving oscillatory flows
	Flow paths		Pressure distributions			Accurately characterize primary flow resistances
Density Lock Performance	Flow rate	NA	Size to event	NA	NA	Partial blockages should be simulated. Total blockage of the density locks may be a beyond design basis event
	Flow distribution					Demonstrate density lock control in a large-scale integral facility Demonstrate density lock stability in a large-scale integral facility

Table B-3.4.1.2 (continued)

**Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for
Representative PIUS Sequences in Event Category-I**

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.1)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Flow Distributions with Multiple Paths	Flow rate	All applicable paths	All applicable paths	All applicable paths	All applicable paths	Boron transport and spatial distribution may be affected by the availability of multiple flow paths Ensure coupling to core neutronic behavior
Mixing Flows	Flow rate Concentration	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	High and low-boron concentration fluid streams mix at several locations including the pump inlets, forced circulation inlet plenum, and core Ensure coupling to core neutronic behavior
Boron Transport and Diffusion	Concentration	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	In addition to prototypic tests, trace step function injections

Table B-3.4.1.2 (continued)

**Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for
Representative PIUS Sequences in Event Category-I**

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.1)				COMMENTS
	CONFIRMING	1	2	3	4	
	PHENOMENA	PARAMETER RANGES OF DATA				
Reactivity Insertions via moderator temperature, boron, and fuel temperatures	Core power including spatial variations	zero to 120%	zero to anticipated multiple of 100%	zero to 120%	zero to 120%	Applicability of test must be demonstrated to ensure that the data obtained adequately represent the core neutronic coupling with the system thermal-hydraulic behaviors
Pool Cooling	Flow rate Temperature Pressure	Size to transient	Size to transient	Size to transient	Size to transient	Identify pool natural circulation flow patterns and impact upon system performance

Table B-3.4.2.1

Integral Data Types Useful for Confirming Key Phenomena in the Area of Physics for Representative PIUS Sequences in Event Category-II

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Core Power	spatial core variations due to moderator temperature, boron concentration, and void	BOC, EOC and mid-range	Worst case	Worst case	Worst case	Dynamic coupling, in contrast to quasi-static coupling, should be demonstrated Normal power peaking factors at transient initiations should be included

Table B-3.4.2.2

Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for Representative PIUS Sequences in Event Category-II

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Pumped Coolant Flows	Flow rates	Primary system flows	Primary and secondary system flows	Primary system flows	Primary system flows	Trips using the scram-valve system require forced circulation
	Flow paths	Pressure distributions	Pressure distributions	Pressure distributions	Pressure distributions	Accurately characterize primary flow resistances and pump performance
Natural Circulation Coolant Flow	Flow rates	NA	NA	Primary system flows	Primary system flows	Make provisions for resolving oscillatory flows
	Flow paths			Pressure distributions	Pressure distributions	Accurately characterize primary flow resistances
				Liquid levels	Liquid levels	
Density Lock Performance	Flow rate	NA	NA	Size to event	Size to event	Demonstrate density lock control in a large-scale integral facility
	Flow distribution					Demonstrate density lock stability in a large-scale integral facility

Table B-3.4.2.2 (continued)

**Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for
Representative PIUS Sequences in Event Category-II**

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
	CONFIRMING	1	2	3	4	
	PHENOMENA	PARAMETER RANGES OF DATA				
Flow Distributions with Multiple Paths	Flow rate	All applicable paths	All applicable paths	All applicable paths	All applicable paths	Boron transport and spatial distribution may be affected by the availability of multiple flow paths Ensure coupling to core neutronic behavior
Mixing Flows	Flow rate Concentration	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	High and low-boron concentration fluid streams mix at several locations including the pump inlets, forced circulation inlet plenum, and core Ensure coupling to core neutronic behavior

Table B-3.4.2.2 (continued)

**Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for
Representative PIUS Sequences in Event Category-II**

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Flow Regime Changes Affecting (a) Surface-Fluid Heat Transfer (b) Density and Buoyancy	Void fraction	NA	NA	Void fraction from 0 to 1	Void fraction from 0 to 1	Make provisions for core uncovery and recovery
	Fluid pressure and temperature			Fluid pressure and temperature	Fluid pressure and temperature	Natural circulation flow rates are a function of core outlet and riser temperature distributions
Boron Transport and Diffusion	Concentration	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	Low level to 2200 ppm	In addition to prototypic tests, trace step function injections
Reactivity Insertions via moderator temperature, boron, void, and fuel temperatures	Core power including spatial variations	Zero to 120%	Zero to needed multiple of 100%	Zero to 120%	Zero to 120%	Applicability of test must be demonstrated to ensure that the data obtained adequately represent the core neutronic coupling with the system thermal-hydraulic behaviors

Table B-3.4.2.2 (continued)

**Integral Data Types Useful for Confirming Key Phenomena in the Area of Thermal-Hydraulics for
Representative PIUS Sequences in Event Category-II**

KEY PIUS THERMAL- HYDRAULIC PHENOMENA	DATA TYPES USEFUL FOR CONFIRMING PHENOMENA	REPRESENTATIVE EC-II SEQUENCES (See Table B.3.2.2)				COMMENTS
		1	2	3	4	
		PARAMETER RANGES OF DATA				
Break Flows	Flow rate	NA	Size to break	Size to break	Size to break	Simulate the siphon breaker
	Void fraction					Break flows are coupled to many other system behaviors enumerated above
	Break orientation					Test a range of break sizes
Pool Cooling	Flow rate	Size to transient	Size to transient	Size to transient	Size to transient	Identify pool natural circulation flow patterns and impact upon system performance
	Temperature					
	Pressure					

Table B-3.5.1.1

Existence of Data Bases in the Area of PIUS Reactor Physics for EC-I Sequences

PIUS REACTOR PHYSICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Reactivity vs moderator temperature	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS
Reactivity vs fuel temperature	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS
Reactivity vs boron concentration	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS
Boron transport across thermal gradients Single tube	Ref. B-1	Partial	Completed. Measured solute transport across the temperature gradient in a single density lock tube under prototypic pressure and temperature conditions. The facility was subjected to sloshing boundary conditions measured in the full scale tests.

Table B-3.5.1.2

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-I Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Stability of thermally stratified layers Small scale	Ref. B-1	Partial	Completed. Demonstrate the stability of such layers
Stability of thermally-stratified layers Full scale	Ref. B-1	Potentially Adequate	Planned. Model hot/cold "sloshing" in the density lock pipes exposed to representative turbulence conditions
Hydraulic stability Reduced scale test of upper density lock and relevant portion of riser	Ref. B-2 Ref. B-3	Potentially Adequate	Planned. Characterize flow distributions. Measure the detailed flow conditions in the riser and characterize the disturbance to the density lock from the reactor coolant flow

Table B-3.5.1.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-I Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Density lock test Full scale	Ref. B-1 Ref. B-3	Potentially adequate	Planned. Tests will (a) characterize the transport mechanisms in the density lock, (b) verify the design features provided to reduce excessive mixing of the primary coolant and pool water in the density lock, and (c) characterize the dynamic behavior of the density lock. the range of disturbances will span those expected in both the lower and upper density locks.
Transient flow conditions in the reactor pool Approximately 1/10th scale test	Ref. B-2 Ref. B-3	Potentially adequate	Characterize transient flow conditions in the reactor pool following release of hot water through the upper density lock after trip of one or more of the reactor coolant pumps.
Reactor coolant pump Full scale	Ref. B-1	Potentially adequate	Unclear whether pump performance will be characterized under two-phase and coastdown conditions
Steam generator Full length tube model	Ref. B-1	Potentially adequate	Test will characterize thermal performance and part-load operation.

Table B-3.5.1.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-I Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Proof-of-principle integral tests	Ref. B-4 to B-7	Adequate for proof-of-principle	Integral tests in two small facilities. Limited scaling rationale and instrumentation. Control and activation of density locks demonstrated
<p>Small-scale integral test.</p> <p>ABB Atom has conducted small-scale, full-height integral tests in a facility named ATLE. The volume scale is 1:308. Facility, simulates many, but not all, key systems, components, processes, and phenomena. Small scale may produce facility atypicalities. Tests may be used to provide a partial benchmarking of codes for design or safety analysis applications.</p>	<p>Ref. B-1 Ref. B-8</p>	Partial	<p>Completed.</p> <p>A limited test program of operational and anticipated transients was conducted.</p>

Table B-3.5.1.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-I Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
<p>Supplementary small-scale integral test</p> <p>The ATLE facility will be modified to better reflect the actual PIUS design in terms of hydraulic arrangement, water flow velocities, and linear power.</p>	Ref. B-3	Partial	<p>Planned. Complement and extend the test program already completed in the ATLE test facility. Cover a spectrum of EC-I events.</p> <p>Some facility atypicalities will continue to exist in the modified facility.</p>
<p>Large-scale integral test</p> <p>The facility is described as full-height and 1/5th diameter.</p> <p>The objective of the test program is to characterize the operational behavior of the complete PIUS system in a large-scale facility</p>	Ref. B-2 Ref. B-3	Ref. B-2 Ref. B-3	Planned. The test program is described as including verification of start-up procedures from cold conditions to full temperature, normal operation including load following and operational transients, and shut-down procedures to hot and cold shutdown conditions
<p>Passive decay heat removal</p> <p>The facility will be full scale in elevation</p>	Ref. B-2 Ref. B-3	Potentially adequate	Planned. Characterize the performance of the passive decay heat removal system including the water cooler immersed in the reactor, the intermediate closed cooling natural circulation system, and the natural draft air coolers

Table B-3.5.2.1

Existence of Data Bases in the Area of PIUS Reactor Physics for EC-II Sequences

PIUS REACTOR PHYSICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Reactivity vs moderator temperature	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS
Reactivity vs fuel temperature	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS
Reactivity vs boron concentration	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS
Reactivity vs void	No PIUS-specific data base	Inadequate	PIUS will use a PWR-type fuel, although of shorter length. ABB may be able to demonstrate that significant portions of this data base are applicable to PIUS

Table B-3.5.2.2

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-II Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Boron transport across thermal gradients Single tube	Ref. B-1	Partial	Completed. Measured solute transport across the temperature gradient in a single density lock tube under prototypic pressure and temperature conditions. The facility was subjected to sloshing boundary conditions measured in the full scale tests.
Stability of thermally stratified layers Small scale	Ref. B-1	Partial	Completed. Demonstrate the stability of such layers
Stability of thermally-stratified layers Full scale	Ref. B-1	Potentially adequate	Planned. Model hot/cold "sloshing" in the density lock pipes exposed to representative turbulence conditions
Hydraulic stability Reduced scale test of upper density lock and relevant portion of riser	Ref. B-2 Ref. B-3	Potentially adequate	Planned. Characterize flow distributions. Measure the detailed flow conditions in the riser and characterize the disturbance to the density lock from the reactor coolant flow

Table B-3.5.2.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-II Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Density lock test Full scale	Ref. B-1 Ref. B-3	Potentially adequate	Planned. Tests will (a) characterize the transport mechanisms in the density lock, (b) verify the design features provided to reduce excessive mixing of the primary coolant and pool water in the density lock, and (c) characterize the dynamic behavior of the density lock. the range of disturbances will span those expected in both the lower and upper density locks.
DNB test for PIUS fuel assembly	Ref. B-3	Potentially adequate	Verify the DNB correlation used in the analysis of normal operation, transients, and accidents.
Transient flow conditions in the reactor pool Approximately 1/10th scale test	Ref. B-2 Ref. B-3	Potentially adequate	Characterize transient flow conditions in the reactor pool following release of hot water through the upper density lock after trip of one or more of the reactor coolant pumps.

Table B-3.5.2.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-II Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
Siphon breaker Full-scale verification of detailed design at atmospheric pressures	Ref. B-1 Ref. B-2 Ref. B-3	Potentially adequate	Verify the hydraulic performance of the siphon breaker during normal operating conditions, including startup and shutdown. Verify operation following primary system, cold-leg breaks.
Reactor coolant pump Full scale	Ref. B-1	Potentially adequate	Unclear whether pump performance will be characterized under two-phase and coastdown conditions
Steam generator Full length tube model	Ref. B-1	Potentially adequate	Test will characterize thermal performance and part-load operation.
Proof-of-principle integral tests	Ref. B-4 to B-7	Adequate for proof-of-principle	Integral tests in two small facilities. Limited scaling rationale and instrumentation. Control and activation of density locks demonstrated

Table B-3.5.2.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-II Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
<p>Small-scale integral test.</p> <p>ABB Atom has conducted small-scale, full-height integral tests in a facility named ATLE. The volume scale is 1:308. Facility, simulates many, but not all, key systems, components, processes, and phenomena. Small scale may produce facility atypicalities. Tests may be used to provide a partial benchmarking of codes for design or safety analysis applications.</p>	<p>Ref. B-1 Ref. B-8</p>	<p>Partial</p>	<p>Completed.</p> <p>A limited test program of operational and anticipated transients was conducted. There has been no testing of more severe transients</p>
<p>Supplementary small-scale integral test</p> <p>The ATLE facility will be modified to better reflect the actual PIUS design in terms of hydraulic arrangement, water flow velocities, and linear power.</p>	<p>Ref. B-3</p>	<p>Partial</p>	<p>Planned. Complement and extend the test program already completed in the ATLE test facility. Include LOCA testing.</p> <p>Some facility atypicalities will continue to exist in the modified facility</p>

Table B-3.5.2.2 (continued)

Existence of Data Bases in the Area of PIUS Thermal-Hydraulics for EC-II Sequences

PIUS THERMAL-HYDRAULICS DATA	EXISTING AND PLANNED DATA BASES	COVERAGE OF OVERALL DATA BASE	COMMENTS
<p>Large-scale integral test</p> <p>The facility is described as full-height and 1/5th diameter.</p> <p>The objective of the test program is to characterize the operational behavior of the complete PIUS system in a large-scale facility</p>	<p>Ref. B-2</p> <p>Ref. B-3</p>	<p>Ref. B-2</p> <p>Ref. B-3</p>	<p>Planned. The test program is described as including verification of start-up procedures from cold conditions to full temperature, normal operation including load following and operational transients, and shut-down procedures to hot and cold shutdown conditions</p>
<p>Passive decay heat removal</p> <p>The facility will be full scale in elevation</p>	<p>Ref. B-2</p> <p>Ref. B-3</p>	<p>Potentially adequate</p>	<p>Planned. Characterize the performance of the passive decay heat removal system including the water cooler immersed in the reactor, the intermediate closed cooling natural circulation system, and the natural draft air coolers</p>

Table B-3.6.1

Prioritized Areas of Missing or Partial Data for the PIUS Reactor in EC-I Sequences

PRIORITY	DATA TYPE	COMMENTS
1	Boron reactivity coefficient	The PIUS reactor does not include mechanical control and shutdown rods. Reactivity and shutdown are effected via the moderator temperature and boron concentration. Although the PIUS fuel is of the PWR type, ABB has not documented, to date, its rationale for not providing this essential data.
2	Moderator temperature reactivity coefficient	Same comment as for priority 1
3	Flow distributions in large-scale component and/or integral tests	Spatial distributions have not been characterized in either the ABB analytical or experimental efforts to date. Flow and temperature spatial variations in the core may lead to unanticipated core neutronic behavior. Some information may be obtained from large-scale, single-component tests but a large-scale integral test is highly desirable.
4	Coupled core neutronic and system thermal-hydraulic behavior	Same comment as for priority 1. In the ATLE test an indirect simulation of these coupled phenomena was used. This approach may also be adequate in a large-scale integral facility but the acceptability of this approach as the only demonstration of the coupled core neutronic and system thermal-hydraulic behavior has yet to be documented.
5	Reactor coolant pump characterization under liquid flow conditions	None of the EC-1 events lead to pump operation under two phase conditions.
6	Flow and heat transfer characteristics of the passive pool cooling system	

Table B-3.6.1 (continued)

Prioritized Areas of Missing or Partial Data for the PIUS Reactor in EC-I Sequences

PRIORITY	DATA TYPE	COMMENTS
7	Boron concentration measurement	See priority 1. It is essential that an accurate, reliable, and timely measurement system for primary and pool boron concentrations exist. Such systems may be off-the-shelf but this has not yet been documented.

Table B-3.6.2

Prioritized Areas of Missing or Partial Data for the PIUS Reactor in EC-II Sequences

PRIORITY	DATA TYPE	COMMENTS
1	Boron reactivity coefficient	The PIUS reactor does not include mechanical control and shutdown rods. Reactivity and shutdown are effected via the moderator temperature and boron concentration. Although the PIUS fuel is of the PWR type, ABB has not documented, to date, its rationale for not providing this essential data.
2	Moderator temperature reactivity coefficient	Same comment as for priority 1.
3	Void reactivity coefficient	Same comment as for priority 1.
4	Flow distributions in large-scale component and/or integral tests	Spatial distributions have not been characterized in either the ABB analytical or experimental efforts to date. Flow and temperature spatial variations in the core may lead to unanticipated core neutronic behavior. Some information may be obtained from large-scale, single-component tests but a large-scale integral test is highly desirable.
5	Coupled core neutronic and system thermal-hydraulic behavior	Same comment as for priority 1. In the ATLE test an indirect simulation of these coupled phenomena was used. This approach may also be adequate in a large-scale integral facility but the acceptability of this approach as the only demonstration of the coupled core neutronic and system thermal-hydraulic behavior has yet to be documented.
6	Core uncover and recovery	Test LBLOCA conditions
7	Reactor coolant pump characterization under liquid flow conditions	None of the EC-1 events lead to pump operation under two phase conditions.

Table B-3.6.2 (continued)

Prioritized Areas of Missing or Partial Data for the PIUS Reactor in EC-II Sequences

PRIORITY	DATA TYPE	COMMENTS
8	Flow and heat transfer characteristics of the passive pool cooling system	
9	Boron concentration measurement	See priority 1. It is essential that an accurate, reliable, and timely measurement system for primary and pool boron concentrations exist. Such systems may be off-the-shelf but this has not yet been documented.

Table B-3.7.1

Prioritized Areas for PIUS Modeling Enhancements for Phenomena of Importance in EC-I Sequences*

PRIORITY	MODELING ENHANCEMENT	COMMENTS
1	Boron tracking	Implement higher-order numerical model to minimize numerical diffusion related to boron. The NRC-approved version of TRAC, version MOD2, does not have this feature. A higher-order Godonov method has been implemented in TRAC-PF1/MOD3 and would be appropriate for the NRC code. Currently, the MOD3 solute tracker does not apply to several component models; the solute tracking capability should be generalized.
2	Provide standard plotting capability for TRAC workstation version.	A TRAC workstation version of the NRC-approved code has been produced and released. However, the plotting capability of the mainframe NRC version is not available on the workstation and this considerably limits the utility of the workstation version.
3	Enhanced graphics display capability	A graphics analyzer capability, like the nuclear plant analyzer, would assist the analyst to perform higher quality and more timely analyses.

* This list is considered to be very preliminary.

Table B-3.7.2

Prioritized Areas for PIUS Modeling Enhancements for Phenomena of Importance in EC-II Sequences^a

PRIORITY	MODELING ENHANCEMENT	COMMENTS
1	Three-dimensional kinetics	At present, the NRC version of the code, MOD2, contains a point kinetics model of core neutronic behavior. Since reactivity control is via moderator temperature and the transport of boron into the core, dimensional effects may be important. Currently, dimensional behavior has not been explored either experimentally or analytically. The NESTLE model developed by North Carolina State University is currently being implemented in the MOD3 version of TRAC and would be appropriate for the NRC code.
2	Boron tracking	Implement higher-order numerical model to minimize numerical diffusion related to boron. The NRC-approved version of TRAC, version MOD2, does not have this feature. A higher-order Godonov method has been implemented in TRAC-PF1/MOD3 and would be appropriate for the NRC code. Currently, the MOD3 solute tracker does not apply to several component models; the solute tracking capability should be generalized.
3	Provide standard plotting capability for TRAC workstation version.	A TRAC workstation version of the NRC-approved code has been produced and released. However, the plotting capability of the mainframe NRC version is not available on the workstation and this considerably limits the utility of the workstation version.
4	Enhanced graphics display capability	A graphics analyzer capability, like the nuclear plant analyzer, would assist the analyst to perform higher quality and more timely analyses.

^a This list is considered to be very preliminary.