Westinghouse Non-Proprietary Class 3

WCAP-16392-NP

July 2005

IRIS (International Reactor Innovative & Secure) Test Plan

Westinghouse

WCAP-16392-NP

IRIS (International Reactor Innovative & Secure) Test Plan

L. E. Conway
Thermal, Fluid, & Nuclear Engineering
Science & Technology Department

July 2005

Approved:
Mario D. Carelli, Chief Scientist
Science & Technology Department

Westinghouse Electric Company LLC
Science and Technology Department
1344 Beulah Road, Building 401
Pittsburgh, PA 15235-5083

LEGAL NOTICE

DOCUMENT AND PAGE REVISION STATUS

Document Identification: IRIS (INTERNATIONAL REACTOR INNOVATIVE & SECURE)

TEST PLAN

Doc. Rev.	0										
Date	07/2005										

Page No.	DOCUMENT REVISIONS											Page No.	DOCUMENT REVISIONS										
	0												0										
	Page Revisions												Page Revisions										

ACKNOWLEDGEMENTS

The Westinghouse IRIS members, Mario Carelli, Luca Oriani, and Bojan Petrovic together with members of the IRIS partner organizations, have provided significant and valuable comments and assistance in defining the IRIS Test Plan.

TABLE OF CONTENTS

a,c

[ ]

TABLE OF CONTENTS (cont.)

LIST OF TABLES

LIST OF FIGURES

LIST OF ACRONYMS

Acronym	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
A-E	Ansaldo-Energia
BDB	Beyond Design Basis
CCB	Configuration Control Board
CCFL	Counter Current Flow Limitation
CFD	Computational Fluid Dynamics
CFR	US Code of Federal Regulations
CHF	Critical Heat Flux
CRDM	Control Rod Drive Mechanisms
CV	Containment Vessel
CVCS	Chemical and Volume Control System
DB	Design Basis
DC	Direct Current
DCD	Design Control Document
DNB	Departure from Nucleate Boiling
DP	Delta-Pressure
DVI	Direct Vessel Injection
EHRS	Emergency Heat Removal System
EBS	Emergency Boration System
EBT	Emergency Boration Tank
EMDAP	Evaluation Model Development and Assessment Procedure
FLB	Feed Line Break
FOAKE	First of a Kind Engineering
FOM	Figure of Merit
GT	Guide Tubes
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
I-CRDM	Internal Control Rod Drive Mechanism
IET	Integral Effects Test
IRIS	International Reactor Innovative and Secure
LBLOCA	Large Break LOCA
LGMS	Long Term Gravity Makeup System
LOCA	Loss of Coolant Accident
MPa	Mega-Pascals
MSLB	Main Steam Line Break
MWe	Megawatt electric
MWt	Megawatt thermal
NRC	Nuclear Regulatory Commission
PIRT	Phenomena Identification and Ranking Table
PCCS	Passive Containment Cooling System

LIST OF ACRONYMS (cont.)

PSS	Pressure Suppression System
PWR	Pressurzied Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RNS	Normal Residual Heat Removal System
RPI	Rod Position Indicator
RV	Reactor Vessel
RVC	Reactor Vessel Cavity
RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SBLOCA	Small Break LOCA
SET	Separate Effects Test
SFWS	Start-up Feedwater System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SiC	Silicon Carbide
SoK	State of Knowledge
T_{hot}	Hot Reactor Water Temperature

ABSTRACT

The purpose of this topical is to provide the test plan for the International Reactor Innovative and Secure (IRIS) nuclear power plant (Reference 1). IRIS is an advanced, integral, light-water cooled reactor of medium generating capacity (335 MWe), geared at near term deployment (2012-2015). IRIS is an innovative design that features an integral reactor vessel that contains all the reactor coolant system components, including the steam generators, coolant pumps, pressurizer and heaters, and control rod drive mechanisms, in addition to the typical core and control rods, and reactor internals. Other IRIS innovations also include a small, high design pressure, spherical steel containment and a simplified passive safety system concept and equipment features that derive from its unique "safety-by-design"™ philosophy. The IRIS "safety-by-design"™ approach not only improves safety, but it also allows a significant reduction and simplification in safety systems.

In order to successfully license the IRIS innovative integral reactor coolant system design, as well as its "safety-by-design"™ approach features, validated and verified safety analyses of the new IRIS equipment and system designs will be required. Therefore, the IRIS design team has developed a test plan that will provide the necessary data for the development, assessment, validation and verification of the computer models used for safety analyses and to confirm the operation of all the IRIS unique features and components. This test plan includes: 1) basic engineering development tests, 2) component separate effects tests [ ]^{a,c}, 3) component separate effects test [ ]^{a,c}, and 4) integral effects tests [ ]^{a,c}. The tests required for design certification will provide thermal-hydraulic data for computer code validation and/or will ensure that new components and system functions important to plant safety are demonstrated.

The IRIS Test Plan has been developed utilizing "The IRIS Small Break LOCA PIRT (Phenomena Identification and Ranking Table)" (Reference 2) for guidance, as well as Westinghouse's recent experience in obtaining design certification of the AP600 plant and final design approval of the AP1000 plant. The IRIS PIRT and the scaling of the actual test articles and facilities are used: [ ]^{a,c} The PIRT and the scaling evaluations are also being presented to the U.S. NRC for review and comment as part of the IRIS pre-certification licensing effort.

1 INTRODUCTION

1.1 BACKGROUND/PURPOSE

Westinghouse Electric Corporation, in conjunction with a large consortium of engineering, academic, and utility organizations is developing an advanced light water reactor design known as IRIS (International Reactor Innovative and Secure). IRIS is a 1000 MWt, 335 MWe, PWR that features an integral reactor coolant system (RCS) and utilizes greatly simplified passive safety systems. The IRIS design goal is to achieve the same overall electrical generation cost as larger nuclear power electrical generation designs while providing an easier and quicker construction, and causing less impact on smaller electric grids. The integral RCS which results in a significant reduction in the size of the containment building size, and the simpler systems which reduce the auxiliary building size, along with modular construction techniques and longer cycle times between refueling shutdowns, are employed to reduce the capital costs, construction time, and operation and maintenance costs.

The IRIS is a new design and, as such, it must conform to the requirements of 10 CFR 50, Part 52, which states, "Certification will be granted only if the performance of each safety feature of the design has been demonstrated through either analysis or the appropriate test programs, experience, or a combination thereof; interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience or a combination thereof; and sufficient data exist on the safety features of the design to assess the analytical tools for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium conditions."

To comply with the requirements of 10 CFR 50, Part 52, the IRIS test plan will provide the necessary data for the development, assessment, and verification of the computer models used for safety analyses. Because the IRIS design contains innovative and new features, the IRIS Test Plan also confirms the manufacturability and operation of all the IRIS unique features and components. This test plan includes: 1) basic engineering development tests, 2) component separate effects tests [ ]^{a,c}, 3) component separate effects test [ ]^{a,c}, and 4) integral effects tests that examine the integrated performance of components and/or systems which are [ ]^{a,c}. The tests required for design certification will provide thermal-hydraulic data for computer code validation and/or will ensure that new components and system functions important to plant safety are demonstrated.

2 IRIS DESIGN SUMMARY AND FEATURES

A brief description of the major features of IRIS is provided below. The reader is referred to Reference 1 for a more complete description of the plant, and References 3 and 4 for results of the IRIS safety analyses.

Integral Reactor Vessel

The IRIS integral reactor vessel (RV) houses not only the nuclear fuel, control rods, and internal structures but also all the major reactor coolant system (RCS) components (see Figure 2-1), including: eight small, spool type, reactor coolant pumps (RCPs); eight modular, helical coil, once through steam generators (SGs); the pressurizer and heaters located in the RV upper head; and, the control rod drive mechanisms. This integral RV arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a more compact configuration and in the elimination of the large loss-of-coolant accident as a design basis event. Because the IRIS integral vessel contains all the RCS components, it is larger than the RV of a traditional loop-type PWR, and it has a unique reactor coolant flow path. Water flows upwards through the core and the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper part of the annular plenum between the extended core barrel and the RV inside wall, where the suction of the RCPs is located. The flow from each of the eight RCPs is directed downward through its associated helical coil steam generator module. The primary flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the flow circuit.

The major primary system components within the integral reactor vessel are briefly described below.

Reactor Coolant Pumps

IRIS has adopted "spool-type" axial flow RCPs that are totally contained inside the RV in order to eliminate the need for a large pump penetration through the reactor vessel structure and the associated pump mounting flange and pump pressure housing.

The IRIS RCP, in its simplest form, consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor that carries a high specific speed pump impeller. Pumps of this type have been used in marine and chemical plant applications where large flow rates at low developed head are needed. Because of their low developed head, spool pumps have never been candidates for nuclear applications. The integral configuration, low pressure drop IRIS can accommodate these pumps and take advantage of these characteristics. However, because the pump is immersed in the hot reactor coolant inside the IRIS reactor vessel, the winding insulation and the bearings must be capable of operating in a ~330°C and ~15.5 MPa water environment.

Therefore, the IRIS RCP sealed, high temperature, stator and rotor windings will be developed, fabricated, and tested. Also, the material used for the hydrostatic, water lubricated bearings will be verified by test, since the IRIS water temperatures are higher than the liquid temperature in current bearing designs. In addition, the bearing design will include features to address the relative growth of the bearing surfaces to maintain the critical bearing clearance. Significant design experience exists in these areas but testing is needed to establish the best design and material selection fit for the IRIS application.

Control Rod Drive Mechanisms

The IRIS CRDMs will also be located inside the reactor vessel, in the riser region directly above the control rod guide assemblies. Locating the CRDMs inside the reactor vessel eliminates the possibility of a control rod ejection accident (a Class IV accident), because there is no potential to create a large differential pressure to drive the CRDM out of the reactor vessel (the drive rod and drive mechanism are both inside the RV). This CRDM location also eliminates the large, thin-walled, sleeved, CRDM drive-rod penetrations in the reactor vessel upper head eliminating problems related to corrosion cracking of these sleeves, the sleeve to clad welds, and sleeve to CRDM pressure housing seals or welds, which have resulted in the periodic inspection and replacement of the reactor vessel upper heads (e.g., the Davis-Besse plant).

The IRIS internal CRDM (I-CRDM) design will be [

][a,c] Note that in the current external CRDM these parts are all exposed to reactor coolant at 15.5 MPa pressure and temperatures as high as 260°C. However, the IRIS I-CRDM will differ from the current design in three main areas:

a,c

The I-CRDM mechanical parts will be assured to be manufactured from materials that are designed to minimize corrosion and cracking at the IRIS core exit temperature.

Helical Coil, Once-Through Steam Generator

The IRIS steam generators (SGs) are a once-through, helical-coil tube bundle design (See Figure 2-2) with the primary fluid outside the tubes and secondary fluid inside the tubes. The tubes are connected to the lower feed water header and the upper steam header walls which act as the tube sheets. The helical-coil, tube bundle design is capable of accommodating thermal expansion without excessive mechanical stress, and the tube inlets are individually orificed to minimize parallel path flow instability.

A prototype of this SG was successfully tested by Ansaldo-Energia (A-E) using a 20 MWt full diameter, part height, test article. This test established the heat transfer, primary and secondary side pressure loss, and parallel flow stability performance characteristics of this type of SG. A key test objective was to determine the stable operating domain for the tested SG arrangement.

Although the above mentioned A-E test program has provided a significant data base for the overall design and operation of the helical-coil, once-through IRIS SG, this test program did not include the full range of conditions that will occur in IRIS reactor operation. Also, the IRIS preliminary design effort has incorporated many new features, design modifications, and manufacturing techniques that will require engineering development testing prior to the construction of a prototypic IRIS SG test article. These IRIS design improvements include:

- Thick walled SG tubes and secondary piping up to the steam and feed water isolation valves that are designed for [ ]^{a,c} differential pressure. This feature not only reduces the likelihood of a SGTR, but allows the SG to be [ ]^{a,c}.

  These thicker walled tubes will impact the manufacturing methods used for the IRIS SGs.

- The tube to tube sheet attachment in the A-E test article utilized non-prototypic welded connections. These welds would be considered Class 1 primary pressure boundary welds and would require periodic volumetric inspection. The inspection of welds in all the SG tubes would be difficult and time consuming, and would represent a significant departure from current SG design practice. In addition, the use of the [ ]^{a,c} for the IRIS SG. This is because [ ]^{a,c}.

  Therefore, the IRIS SG design will incorporate [ ]^{a,c}

- The IRIS SGs are not only utilized for normal steam generation operation - they also operate in conjunction with the IRIS EHRS, to provide safety-related heat transfer from the primary system following postulated accidents and anticipated transients. This results in the IRIS SGs having an operating domain (primary and secondary pressures and temperatures) that is much broader than the domain tested by A-E.

Small Spherical Containment

Because the IRIS integral RV configuration eliminates the loop piping and the externally located steam generators, pumps and pressurizer with their individual vessels, the IRIS containment system is greatly reduced in size. This size reduction, combined with the spherical geometry, results in a design pressure capability at least three times higher than a typical loop reactor cylindrical containment, assuming the same metal thickness and stress level in the shell. The current layout features a spherical, steel containment vessel (CV) that is 25 meters (82 feet) in diameter. The CV is constructed of 1 ¾" steel plate and has a design pressure capability of 1.4 MPa (~190 psig). The containment vessel has a bolted and flanged closure head at the top that provides access to the RV upper head flange and bolting. Refueling of the reactor is accomplished by removing the containment vessel closure head, installing a sealing collar between the CV and RV, and removing the RV head. The refueling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refueling cavity. Fuel assemblies are vertically lifted from the RV directly into a fuel handling and storage area, using a refueling machine located directly above the CV. Thus, no refueling equipment is required inside containment and the single refueling machine is used for all fuel movement activities.

The IRIS containment features a pressure suppression system that limits the containment peak pressure to well below the CV design pressure. The suppression pool water is contained in redundant, independent pools and is located such that it provides a source of elevated water for gravity driven makeup to the RV. Also, the containment is constructed with a RV flood-up cavity in which the lower portion of the reactor vessel is located. This flood-up cavity ensures that the lower section of the RV, where the core is located, is surrounded by water following any postulated accident. The water flood-up height is sufficient to provide long-term gravity makeup, so that the RV water inventory is maintained above the core for an indefinite period of time. It also provides sufficient heat removal from the RV external surface to prevent any vessel failure following beyond design basis core damage scenarios.

Passive Safety Systems

IRIS employs simple and effective passive safety systems that mitigate the effects of postulated design basis events. Following postulated small break LOCAs, these systems minimize the loss of coolant, depressurize the containment, and provide long-term core cooling by RV makeup. A schematic of the IRIS passive safety systems is shown in Figure 2-3 and includes:

Passive emergency heat removal system (EHRS) – which connects to the main steam and feed water piping and operates in natural circulation, removing heat from the primary system through the steam generator heat transfer surface, condensing steam in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. EHRS operation condenses the steam produced by the core directly inside the reactor vessel thus depressurizing the RV while minimizing the loss of mass from the primary system and transferring the decay heat to the environment. The EHRS is sized so that a single subsystem can provide core decay heat removal in the case of a loss of secondary system heat removal capability.

Emergency boration system (EBS) – which provides a diverse means of reactor shutdown by delivering borated water to the RV at any pressure using only gravity. EBS operation also provides a limited amount of makeup water to the primary system.

Small automatic depressurization system (ADS) – vents steam from the pressurizer steam space in order to assists the EHRS in depressurizing the reactor vessel. This ADS consist of parallel, redundant, 4-inch lines, each with two normally closed valves. A single ADS line downstream of the closed valves discharges the vented steam into a water tank through a sparger.

Long term gravity makeup system (LGMS) – consists of piping and valves which enable gravity makeup of water to the RV from the flooded lower containment. This makeup ensures that core cooling is maintained for an unlimited time following any design basis LOCA.

As in the AP600/AP1000, the IRIS safety system design uses natural gravitational forces instead of active components such as pumps, fan coolers or sprays, and their supporting systems.

Additional information on IRIS design, performance, and safety features is contained in References 3 and 4.

2.1 IRIS TEST CLASSIFICATION AND BASES

In order to identify a comprehensive test program to support design certification, the IRIS design must be reviewed from several different aspects to assess not only the safety related functions, but also the manufacturability, proper operation, reliability, inspectability, and maintainability of components. The following issues were considered in the development of the IRIS test program:

1. Components which are new and require engineering development, new materials, new fabrication techniques, or component qualification for reactor use.

2. Safety related systems that are different from the AP600/1000 and standard Westinghouse plants and therefore require testing to demonstrate their operating characteristics.

3. Thermal-hydraulic phenomena that play an important role in the functioning of the IRIS passive safety systems and for which existing experimental data may not be sufficient.

4. Data needed to develop computer models of the IRIS passive safety system thermo-hydraulic phenomena.

5. Applicable data which capture key IRIS phenomena, already in existence for code validation, such as existing Westinghouse data or data in the public domain (e.g., NUREG-1230).

In order to identify the thermal-hydraulic phenomena that play an important role in the function of the IRIS passive safety systems, and to identify the data needed to develop and verify computer models for accident analyses, Westinghouse organized an expert panel to develop a Phenomena Importance Ranking Table (PIRT) for IRIS (Reference 2). The panel members were carefully selected to insure that the PIRT results reflect internationally recognized experience in reactor safety analysis, and were not biased by program preconceptions internal to the IRIS Program. The primary objective of the IRIS PIRT panel was to identify the relative importance of phenomena in the IRIS response to SBLOCAs. This relative importance, coupled with the current relative state of knowledge for the phenomena, provided the framework to plan the IRIS experimental efforts. The influence of the IRIS PIRT on the IRIS Test Plan is summarized in Section 6 of this report. The SBLOCA was identified as the postulated design basis event where the IRIS response is drastically different from loop-type PWRs. In fact, in IRIS, the SBLOCA break flow is minimized and actually reversed by equalization of the reactor vessel and containment vessel pressure. This pressure equalization eliminates the need for high pressure water injection to the core. This SBLOCA response includes the function of all the IRIS passive safety features. In contrast, the IRIS response to other anticipated design basis events is similar to, or even simplified, compared to the AP600/AP1000 due to the IRIS safety by design approach, and therefore a PIRT is not necessary for test plan development.

The safety functions of each component during the SBLOCA mitigation scenario were examined as to their predictability using the current analysis methods and the methods proposed for use in the IRIS design certification. The differences between the IRIS safety feature functions and existing PWR and AP600/1000 designs are also considered, since the basis for the safety analysis methods is from these current PWR designs. A detailed PIRT has been developed in this process to identify all the important thermal-hydraulic phenomena, and the existing verification for the safety analysis codes is assessed against the current verification for the code, as well as the applicability of the data verification for the IRIS design. The assessment indicates which models require additional verification for the IRIS-specific geometry and conditions. The assessment also indicates which phenomena are of most importance for representing the passive features of the IRIS safety systems.

The engineering data needs are also identified for specific components that are different or unique to the IRIS design. Data that can be obtained through tests will be factored into the safety analysis needs for test design and operation. Those engineering needs not covered by the thermal-hydraulic tests identified for the computer code validation result in the performance of additional separate tests. In some cases, data from existing plants are used to help address the need.

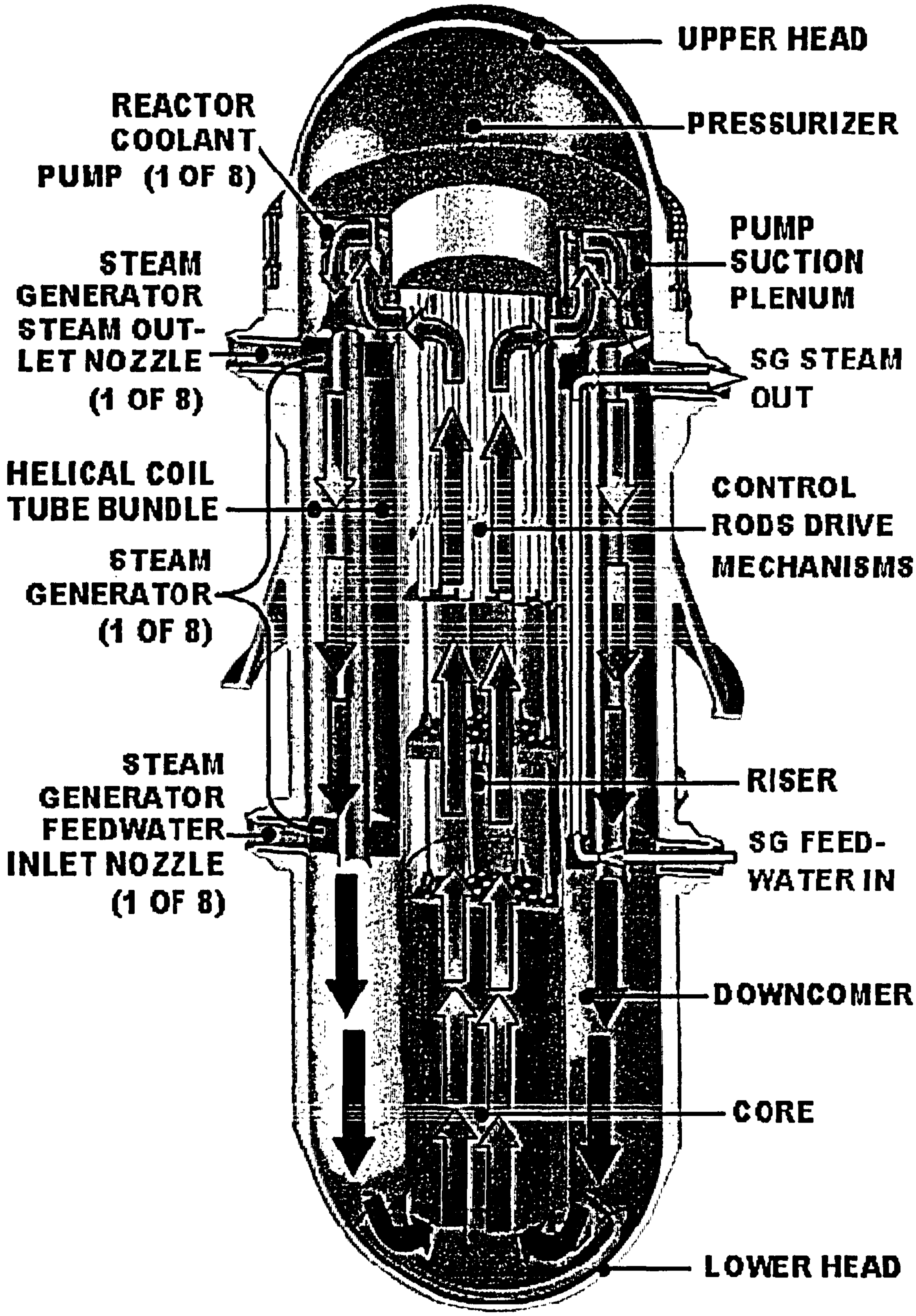

Figure 2-1 IRIS Integral Reactor Vessel/Reactor Coolant System

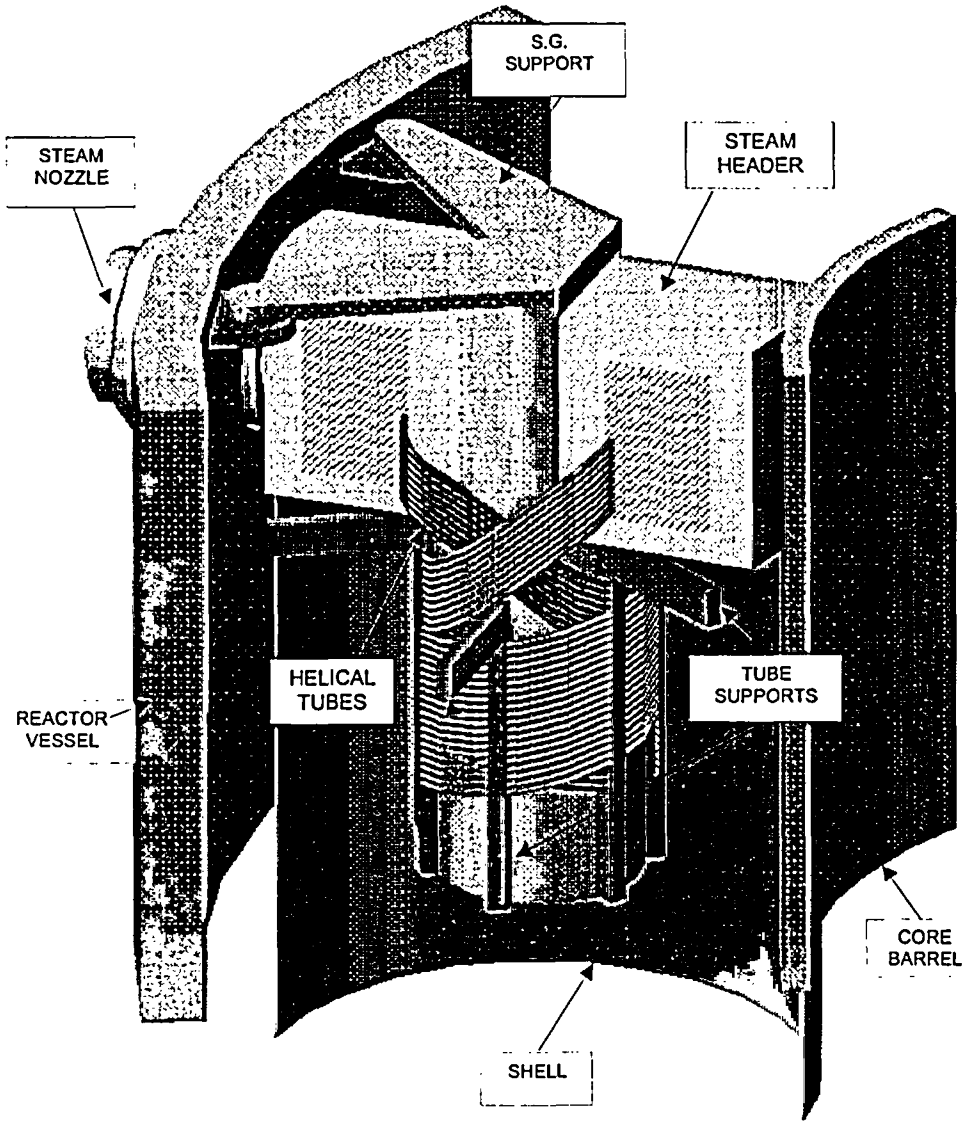

Figure 2-2 IRIS Steam Generator Illustration

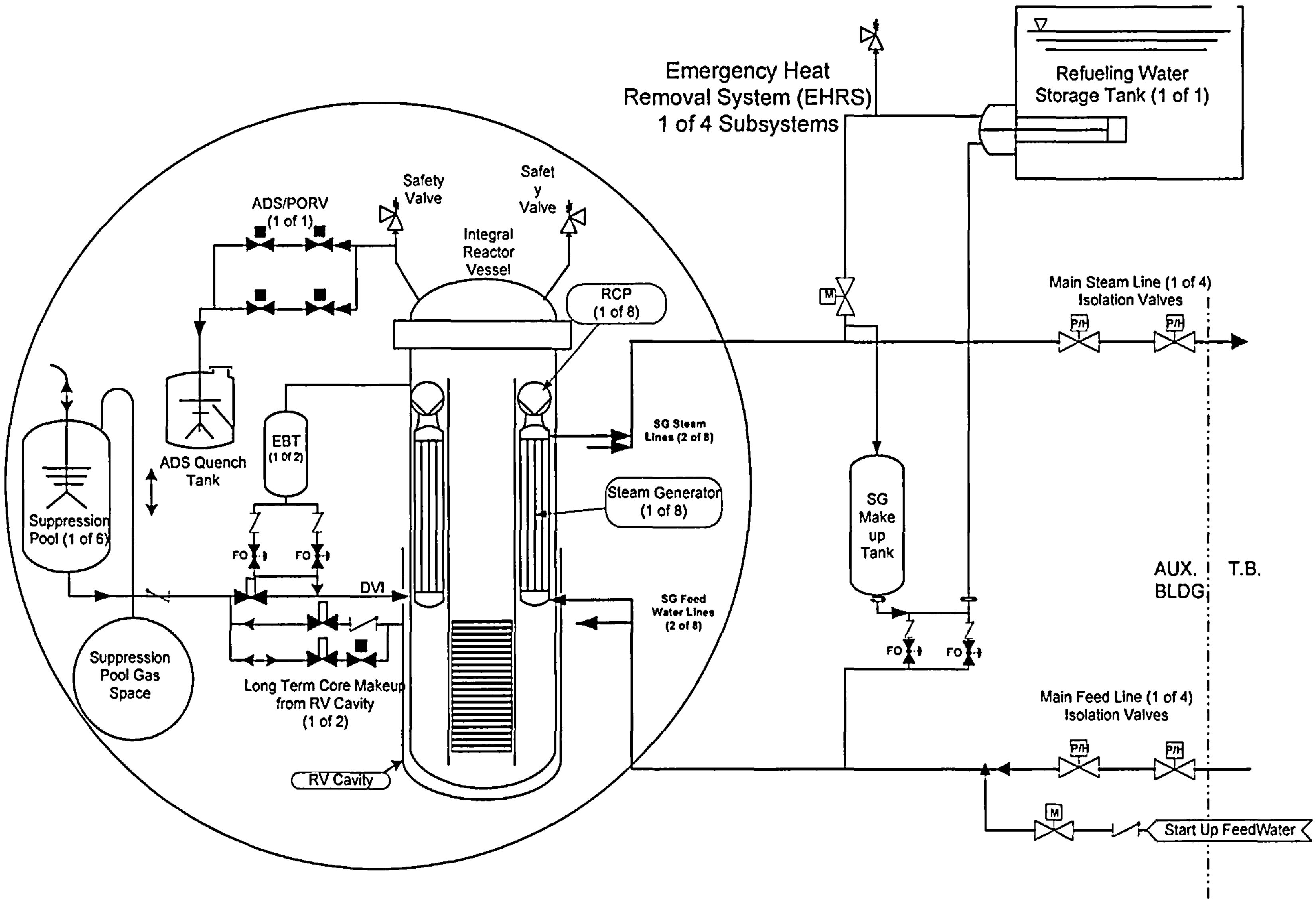

Figure 2-3 IRIS Passive Safety System Schematic

3 IRIS TEST AND ANALYSIS PROCESS

The IRIS test program is aimed at providing both design information and data for computer code validation. The design information will support the IRIS reactor plant goals for simplicity, reliability, safety, and both utility and public acceptance. The data for computer code validation, which will support licensing review and eventual design certification for the IRIS plant by the NRC, must be obtained using established procedures and quality assurance oversight for the entire testing process. This testing process is similar to that used successfully for the design certification of the AP600 and is described below.

An overview of the process that will be used to develop and perform a test in the IRIS test program is presented. A flow diagram that illustrates the test and analysis process used for the IRIS test program is presented in Figure 3-1. The incorporation of the test results into the computer codes used to perform the plant safety analysis is also discussed.

3.1 TEST SPECIFICATION

The testing process will be initiated by the development of a test specification document which describes the objectives of the test and delineates the requirements for the test. These requirements include the overall facility support requirements, a description of the required test article(s), test operational requirements, data acquisition requirements, data reporting requirements, and quality assurance requirements.

Input to the test specification is provided by various functional groups that will ultimately use the data. These inputs include requirements specified by:

- System design – overall system requirements and specifications
- Safety analysis – data needs for computer code validation
- Equipment engineering – component sizes and descriptions
- Licensing – NRC requirements and needs

The test specification is developed and written by the IRIS personnel who will be responsible for overseeing the design and conduct of the test. This responsible person provides the focal point for integrating the needs and requirements of the test facility/component to the data end-users; he is responsible for coordinating the review of the test specification, addressing any identified concerns, resolving comments, and obtaining the appropriate review signatures.

3.2 PRELIMINARY TEST MATRIX

During the development of the test specification document, a draft test matrix is also developed. The parameters of this draft test matrix are required to develop the facility and test article requirements and to determine an envelope for operating the test facility. The draft test matrix is incorporated into the test specification document and identified as preliminary. This preliminary test matrix provides guidance during the design and construction of the test facility.

3.3 SCALING ANALYSIS

For tests that are not full-scale or which employ non-prototypic fluids or fluid conditions, a scaling rationale is developed to support the test specification. The depth of the scaling rationale or analysis performed varies for each of the IRIS test programs, however, detailed scaling analysis will be performed using the scaling methodology of NUREG/CR-5809, "An Integrated Structure and Scaling Methodology for Severe Accident Technical Resolution, Appendix D: A Hierarchical Two-Tiered Scaling Analysis." The scaling results are, of course, required input for the design of the test facility and test articles, and also are used to further develop the test matrix. Therefore, a scaling report delineating the results of the scaling analysis is issued prior to finalizing the test design.

3.4 PRE-TEST ANALYSIS

Analysis of the performance of the test facility is completed to support the development of the test specification document, test matrix, and design of the test facility. Preliminary analyses are performed to establish the test initial and boundary conditions and the facility requirements, e.g., steam and water flow rates, power requirements, etc. These analyses are also used to establish the operating conditions of the facility and the range of conditions that will occur for each matrix test run. These results are input into engineering and system design analyses to confirm the conceptual test facility designs and to evaluate alternative designs.

After the test facility is constructed, the results of the pre-operational tests (described below) are used to refine further the analytical models of the test facility. Actual facility as-built characteristics (including measured volumes and flow pressure losses, heat capacity, and heat losses to the surrounding environment) are used as input to, or boundary conditions for, the pre-test analysis computer code model. This refined model is used to perform pre-test predictions for the matrix tests. The pre-test predictions are used to verify the test operating procedures to establish that the operating conditions during testing are within the limits of the facility/component operating limits and design parameters.

3.5 PRE-OPERATIONAL TESTING

Prior to the initiation of the matrix tests, a series of pre-operational tests are performed to satisfy a number of objectives, including:

- Verify the facility operates as designed
- Check instrumentation and data acquisition system (DAS)
- Provide training for facility operators
- Obtain as-built data for facility components and systems
- Obtain data on facility performance characteristics, e.g., heat losses and piping resistance

During facility construction, components and systems are checked and tested as they are installed or assembled, whenever possible. Prior to testing, the facility/components are hydro and leak tested, as required, and instrumentation and hookups to the DAS are inspected and tested. Final verification of the instrumentation, control, and DAS functions are performed and verified during the pre-operational test phase.

Cold pre-operational tests are those tests that can be performed with the facility at ambient temperature (non-heated) and in most cases unpressurized. These tests include the traditional "shakedown" tests performed to verify operational readiness of the facility, and tests that provide information on the characteristics of the facility. Tests are performed to verify component actuations and proper operation, support system operations, and to fill and drain the facility/components to measure as-built volumes. Flow tests are performed to verify pump operation and to measure pressure drops to obtain line resistances.

Hot pre-operational or hot functional tests are performed to verify the operation of the entire test facility over its full pressure and temperature operating range. A series of tests are completed to confirm the test facility and individual component operating characteristics, to verify the operation of all instrumentation and control capabilities, and to measure the facility and component heat losses.

The cold and hot pre-operational testing provides data for use in the test data reduction plan, and these test results are reported to all interested data users for their review and comment. These data are also used to update the pre-test analysis model to further refine the matrix test predictions and to eliminate or reduce uncertainties when analyzing the data obtained during matrix testing by using measured rather than calculated values (e.g., [ ]^{a,c}.

Quality assurance controls are applied to cold and hot pre-operational tests in the same manner as that of the matrix tests in order to use the data obtained from these tests in the code validation process.

3.6 MATRIX TESTS

Matrix testing is started after verifying that the facility meets the requirements of the test specification, that the facility operates properly, and that the instrumentation is properly installed and calibrated. The preliminary test matrix is finalized as a result of the pre-operational tests, pre-test analysis, and other issues or needs identified from reviews of the initial pre-operational test data reduction and test report.

The objective of matrix testing is to provide accurate, reproducible, high-fidelity data to the various functional groups (identified in Subsection 3.1), that meet the quality assurance requirements in the test specification document. To satisfy this objective, acceptance criteria are established to determine the conditions for test acceptance.

The test acceptance is generally performed at the test site by the responsible test engineer who makes a determination as to whether the test met pre-established, minimum acceptance criteria (operability of key instruments and components, actual versus specified initial conditions, etc.). If a test is deemed unacceptable, the data are not used, and the test is rerun. This review is delivered in the Day-of-Test Report for each test run. If the test is considered valid, data reduction proceeds, and a test run data report is generated.

3.7 DATA REPORTS

Acceptance criteria are also established to determine the conditions for acceptability for the test data as part of the data reduction and data report. These test validation criteria are related to instrumentation operability requirements, initial test condition specifications and tolerances, operating procedures, mass balance, power decay, and review of test instrumentation responses following test completion. Tests that meet the acceptance criteria are used in the computer code validation process. Although tests that did not meet the acceptance criteria are not used for code validation, the test data are maintained to complete the test record. Test data are collected and reported to the data users in a two-step process. The first data report, termed Quick Look Report, is generated to provide access to the test data shortly after the test. The Quick Look Report includes a brief overview of the test(s) performed and summarizes the results of the test(s). Data presented in the Quick Look Report are reviewed against the established test acceptance criteria but are considered preliminary until the final data report is issued.

A Final Data Report is written at the conclusion of the test program to finalize data validation, summarize test results, and assimilate all valid tests and key facility information into a single report. Test data contained in the Final Data Report are reviewed further and validated against the data in the Quick Look Report. Data from the tests are plotted against each other to identify the various parametric effects from the tests. A data error analysis is also provided in the Final Data Report.

3.8 DATA ANALYSIS

Results of the code comparisons to the test data and the necessary code model improvements are documented in a Test Analysis Report. This report documents the test validation and the process of determining whether the overall test was valid via performance of a [ ]^{a,c}. The data contained in the data reports are used as input to the comparisons between the test results and post-test analysis of the tests. Test facility responses such as pressure, temperature, mass distribution, and energy distribution are compared to the computer code results. If warranted, the computer code modeling is improved to better match the test data. The process of checking that the computer code models are correctly solving and applying the respective equations and correlations, including determination of whether the products of a given phase of the software development cycle fulfill the requirements established, is termed code verification. Code validation is the process of checking that the computer code models are correctly predicting the associated phenomena. For IRIS, this involves comparing computer code model results with the validated test data and includes documentation that software test results are correct and consistent with the software functional specifications.

The separate effects tests [ ]^{a,c} are used primarily to obtain data that are used to develop, refine, and, subsequently, verify the computer code models for these key components. The integral systems tests [ ]^{a,c} are used to [ ]^{a,c} as shown in Figure 3-1.

Some of the matrix test runs from the [ ]^{a,c}

[

]a,c

The computer code model is subsequently "frozen" so that no further modifications are made. The model is then used to predict the response of the IRIS plant for the various postulated transients.

3.9 TECHNICAL INTEGRATION AND MANAGEMENT CONTROL

Throughout the test and analysis process, controls are utilized to ensure consistency between design, tests, and analysis and to ensure that the objectives of the program are achieved.

A design change control process will be established to evaluate and approve design changes to the IRIS plant and to implement changes in the test and analysis programs once plant design changes are approved. Each proposed plant design change is submitted for review by the Configuration Control Board (CCB). A description of the change is forwarded for review, comment, and impact evaluation by the system, test, and analytical functional groups. The CCB reviews the evaluations and, if approved, the affected functional group implements the required modifications to the test facilities or performs the necessary analysis and issues the appropriate reports documenting the evaluation of the design change.

a,c

Figure 3-1 IRIS Test and Analysis Process Flow Chart (Page 1 of 3)

a,c

Figure 3-1 IRIS Test and Analysis Process Flow Chart (Page 2 of 3)

a,c

Figure 3-1 IRIS Test and Analysis Process Flow Chart (Page 3 of 3)

4 IRIS TESTING PLAN

The following key elements in the test program have been identified:

a,c

[

•

•

•

•

•

•

]

4.1 IRIS TEST PLAN OUTLINE

The tests to be performed as part of the IRIS design development and design certification are divided into four types according to the scope and primary purpose. These four test types are:

4.1.1 Basic Engineering Development Tests

The basic engineering development tests are used to provide [

]a,c that are needed for analyses or other design certification activities.

Table 4-1 provides a listing of the IRIS Basic Engineering Development Tests. Note that these tests are typically [

]a,c component.

4.1.2 Component Separate Effects Tests [ ]a,c

Component separate effects tests (SETs) are performed to [

]a,c

Additionally, other tests are to be conducted to demonstrate the [

]a,c component.

Table 4-2 provides a listing of the Component Separate Effects Tests that will be [

]a,c

4.1.3 Component Separate Effects Tests [ ]a,c

These component separate effects tests (SETs) will be performed [

]a,c. Table 4-3 provides a list of the Component Separate Effects Tests [
]a,c

4.1.4 Integral Effects Tests

The integral effects tests (IETs) examine [

]a,c Table 4-4 provides a listing of the IRIS Test Plan Integral Systems Effects Tests.

Table 4-1 IRIS Test Plan Basic Engineering Design Tests a,c

Table 4-2 IRIS Test Plan Separate Effects Component Tests (Required for EMDAP and/or Design Certification)

a,c

Table 4-3 IRIS Test Plan Separate Effects Component Tests (for Qualification and Operability)

a,c

Table 4-4 IRIS Test Plan Integral Systems Effects Tests (Required for EMDAP and Design Certification)

a,c

5 TESTS DESCRIPTIONS

The individual tests that will comprise the IRIS Test Program are described in this section. The tests descriptions are grouped according to their type; i.e., basic engineering development, component separate effects [ ]^{a,c}, component separate effects [ ]^{a,c}, or integral systems effects tests.

5.1 BASIC ENGINEERING DEVELOPMENT TESTS

The basic engineering development tests required for IRIS are directed at [

]^{a,c}.

The individual basic engineering development tests for these components which were listed in Table 4-1 are described below.

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

5.2 COMPONENT SEPARATE EFFECTS TESTS [ ]^{a,c}

The IRIS component separate effects tests that will be performed [ ]^{a,c} which were listed in Table 4-2, are briefly described below.

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

5.3 COMPONENT SEPARATE EFFECTS TESTS [

]a,c

The component separate effects tests that will be performed in order to [

]a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

a,c

5.4 INTEGRAL EFFECTS TESTS

Two IRIS integral tests are to provide data on the integrated function of components and systems to perform their safety function(s). These data are [

]^{a,c}. These tests were listed in Table 4-4 and are described below.

a,c

a,c

a,c

a,c

a,c

6 PIRT FINDING INFLUENCE ON THE IRIS TEST PLAN

The primary objective of the IRIS SBLOCA PIRT (Reference 2) was to identify the relative importance of phenomena in the IRIS response to SBLOCAs. This relative importance, coupled with the current relative state of knowledge for the phenomena, provides a framework for the planning of required experimental and analytical efforts. The IRIS Test Plan [

]a,c.

6.1 PIRT OVERVIEW

The PIRT process attempted to identify the needed experimental and analytical database development in a hierarchical sequence. In this context "safety significance" denotes the combination of how influential a behavior may be on a successful mitigation of an accident scenario and how well that behavior may be predicted via experimental data and/or analytical modeling. From this perspective it follows that behaviors of "highest" safety significance are those that have the most influence on the plant response and are also the least well understood and/or predicted with the current state of knowledge. The range of safety significance determinations then progresses down to the most influential behaviors that are moderately well predicted, to moderately influential behaviors that are least well predicted, and finally moderately influential behaviors that are moderately well predicted.

As discussed in Reference 2, the initial IRIS PIRT effort was directed to SBLOCAs because the IRIS response to this event [

]a,c The IRIS PIRT focused on two limiting SBLOCA scenarios: 1) the break of a RV piping connection at low elevation, i.e., one of the 2" DVI (Direct Vessel Injection) lines; and 2) the break of a RV piping connection located at a high elevation on the RV, i.e., one of the 4" CVCS (Chemical And Volume Control/Normal RHR System) lines. Note that these break sizes correspond to the largest primary fluid piping connections to the IRIS reactor vessel.

6.2 PIRT SCENARIO TIME PHASES

The PIRT process also partitioned these two scenarios into logical time phases in which the phenomenological behaviors are reasonably consistent during a phase. These scenario time phases are described in Table 6-1.

6.3 PIRT RANKING OF THE RELATIVE IMPORTANCE OF IDENTIFIED PHENOMENA

The ranking of phenomena relative importance is based on the SBLOCA safety criteria of interest (Figures of Merit). As described in Section 2.2.5 of Reference 2, they are: [

]a,c.

The identified phenomena were ranked as shown in Table 6-2.

For all phenomena ranked as having [

]a,c.

6.4 IRIS TEST TO ADDRESS PIRT FINDINGS

This review of the PIRT and the assigning of IRIS planned testing to address the phenomena is summarized in the following tables:

a,c

6.5 ADDITIONAL PIRT INFLUENCE COMMENTS

The above tables which highlight the important phenomena will [

]a,c

a,c

In addition to the detailed phenomena identification and ranking process outlined in the above listed tables, the IRIS PIRT Panel concluded that continued experimental data and analytical tool development in the following areas were important to satisfy the safety analysis and licensing objectives of the IRIS Program:

a,c

Table 6-1 IRIS SBLOCA Scenario Description

a,c

Table 6-2 IRIS SBLOCA Phenomena Ranking Scale

a,c

Table 6-3 IRIS Planned Test to Address First Tier PIRT Results

[ ]a,c

a,c

Table 6-4 IRIS Planned Tests to Address Second Tier PIRT Results

[ ]^{a,c}

a,c

Table 6-5 IRIS Planned Tests to Address Third Tier PIRT Results

[ ]a,c

a,c

Table 6-6 IRIS Planned Tests to Address Fourth Tier PIRT Results

[ ]a,c

a,c

7 SUMMARY AND CONCLUSIONS

Although much of the IRIS passive safety concept relies on the same principles as previous Westinghouse passive reactor designs, IRIS has incorporated new and innovative features. The IRIS team members recognize the importance of verifying these features by dedicated testing. Therefore, where data on key features are lacking, tests will be designed to evaluate and confirm the engineering and operation of the IRIS components and the overall concept performance.

A plan for an integrated test program has been developed for IRIS for review by the NRC as part of the IRIS pre-certification process. Four classifications of tests are described in the program, i.e., engineering development tests, separate effects component tests [ ][a,c], separate effects component tests [ ][a,c], and integral effects tests. The separate effects component tests [ ][a,c] and the integral effects tests [ ][a,c].

The two primary objectives of the test program are: [ ][a,c]

The necessity for the tests was derived from a PIRT review of the IRIS design which identified the expected thermal-hydraulic phenomena that the IRIS safety analysis computer codes will have to model and calculate with confidence. Also, a review of the design differences between IRIS and the current industry data base was performed which included testing for the AP600/AP1000 passive safeguards plant design and existing PWRs.

All the IRIS safety features will be tested either [ ][a,c]. These data, along with the planned analysis effort, form a comprehensive program that should result in a successful licensing review and approval of the IRIS design.

8 REFERENCES

1. *IRIS Plant Description Document*, WCAP-16062-P and WCAP-16062-NP, Rev. 0, March 21, 2003

2. *IRIS Small Break LOCA Phenomena Identification and Ranking Table (PIRT)*, WCAP-16318-P and WCAP-16318-NP, Rev. 0, August 2004

3. *IRIS Preliminary Safety Assessment*, WCAP-16082-P and WCAP-16082-NP, Vol. I, Rev. 0, July, 2003

4. *IRIS Preliminary Safety Assessment*, WCAP-16082-NP, Vol. II, Rev. 0, October, 2003

5. Draft DG-1120, "Transient and Accident Analysis Methods," June 2002 (available on NRC PERR at accession number ML003770849).

6. F. H. Ruddy, A. R. Dulloo, J. G. Seidel, F. W. Hantz, and L. R. Grobmyer, "Nuclear Reactor Power Monitoring Using Silicon Carbide Semiconductor Radiation Detectors," Nuclear Technology 140, 198 (2002)

7. F. H. Ruddy, A. R. Dulloo, and J. G. Seidel, "Study of the Radiation Resistance of Silicon Carbide Radiation Detectors," Transactions of the American Nuclear Society, 90, 348 (2004)

8. B. Petrović, F. H. Ruddy and C. Lombardi, "Optimum Strategy For Ex-Core Dosimeters/Monitors In The Iris Reactor," in *Reactor Dosimetry in the 21st Century*, J. Wagemans, H. A. Abderrahim, P. D'hondt, and C. De Raedt (Eds.), World Scientific, London (2003) pp 43-50.