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IRIS

INTERNATIONAL REACTOR INNOVATIVE AND SECURE

FINAL TECHNICAL PROGRESS REPORT

November 3, 2003

Principal Investigator: Mario D. Carelli
Westinghouse Electric Company, LLC

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INTERNATIONAL REACTOR INNOVATIVE AND SECURE
(ORIGINAL GRANT KNOWN AS: THE SECURE TRANSPORTABLE AUTONOMOUS LIGHT WATER REACTOR--STAR-LW)

U. S. Department of Energy
Nuclear Energy Research Initiative
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Foreword

This report is required as the final documentation of the work performed under the three-year NERI grant.

The IRIS Consortium is continuing the project after the end of the NERI program with the purpose of commercially deploying IRIS in the next decade.

The document should be therefore seen as a "progress" rather than "final" report. It documents the project activities over a four-year period (October 1999-October 2003).

As in previous reports, documented here is the total work performed under the NERI grant as well as in-house consortium contributions.

Mario D. Carelli on behalf of the IRIS project partnership

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EXECUTIVE SUMMARY

This NERI project, originally started as the Secure Transportable Autonomous Light Water Reactor (STAR-LW) and currently known as the International Reactor Innovative and Secure (IRIS) project, had the objective of investigating a novel type of water-cooled reactor to satisfy the Generation IV goals: fuel cycle sustainability, enhanced reliability and safety, and improved economics. The research objectives over the three-year (1999-2002) program were as follows:

- First year: Assess various design alternatives and establish main characteristics of a point design
- Second year: Perform feasibility and engineering assessment of the selected design solutions
- Third year: Complete reactor design and performance evaluation, including cost assessment

These objectives were fully attained and actually they served to launch IRIS as a full fledged project for eventual commercial deployment. The program did not terminate in 2002 at the end of the NERI program, and has just entered in its fifth year. This has been made possible by the IRIS project participants which have grown from the original four member, two-countries team to the current twenty members, nine countries consortium. All the consortium members work under their own funding and it is estimated that the value of their in-kind contributions over the life of the project has been of the order of $30M. Currently, approximately 100 people worldwide are involved in the project. A very important constituency of the IRIS project is the academia: 7 universities from four countries are members of the consortium and five more US universities are associated via parallel NERI programs. To date, 97 students have worked or are working on IRIS; 59 IRIS-related graduate theses have been prepared or are in preparation, and 41 of these students have already graduated with M.S. (33) or Ph.D. (8) degrees.

This “final” report (final only as far as the NERI program is concerned) summarizes the work performed in the first four years of IRIS, from October 1999 to October 2003. It provides a panoramic of the project status and design effort, with emphasis on the current status, since two previous reports have very extensively documented the work performed, from inception to early 2002.

After a series of trade-off studies, a conceptual design of IRIS was formulated and completed. The preliminary design is currently underway and has progressed to a point where the defining design characteristics are frozen. They are:

- The reference IRIS size is set at 1000 MWt (~ 335 MWe), however the same design configuration covers the 100-335 MWe range with only modest changes in dimensions. The core design features a 4.95% enriched UO_2 fuel in a 17x17 square array assembly, very similar to standard Westinghouse PWR assemblies. The fuel enrichment and projected burnup are within current limits and therefore it presents **no** licensing issues. The IRIS core is, however, designed to be able to accept various configurations (8-year straight burn, with 8-10% fissile UO_2 or MOX fuel; 4-year straight burn with 4.95% enriched UO_2; two and three batch with 4.95% enriched UO_2 and higher burnup). The latter are for initial deployment, while the former can be considered for future reloads.

- The IRIS vessel includes eight helical steam generators, eight “spool-type” pumps, neutron reflector, pressurizer and control rod drive mechanisms. Six different steam generator designs were evaluated and the Ansaldo helical design was chosen both for its performance and for the fact that it had already been extensively tested in a 20 MWt mockup. The fully internal pumps are based on a design developed for chemical applications. They can be operated in a high temperature environment and have large coastdown and run-out capabilities, but have to be qualified for nuclear applications. The pressurizer is of the steam type and the ratio of its volume to reactor power is much larger than loop PWRs, thus allowing very smooth pressure control, and requiring no sprays. The annular space between the neutron reflector and the vessel is wide enough to reduce the fast fluence on the reactor vessel below 10^{14} n/cm^2, as well as the radiation field at the vessel outer surface to the order of 10^{-4} Sv/hr. This has very positive implications for operational and maintenance doses, for long vessel life, as well as for decommissioning and disposal (the “cold” vessel can act as a sarcophagus for the whole reactor internals minus the fuel). The control rods drive mechanisms are located inside the vessel, thus eliminating the vessel head penetrations and associated problems like the ones recently experienced at the Davis-Besse plant. Also eliminated is the potential for control rods ejection.

- The concept of “safety by design” (to physically prevent accidents from occurring rather than coping, by active or passive means, with their consequences) has been developed and articulated in detail and its implementation has widely exceeded expectations. Not only are large LOCAs eliminated from occurring, as can be expected with all integral designs, but a patented containment design has practically eliminated also small and medium LOCAs as a safety concern. In fact, the core remains fully covered, without any safety injection or water makeup. This is made possible by a design which thermo-hydraulically couples the vessel and the containment so that the pressures inside the vessel and the outside pressure in the containment quickly equalize after the pipe breaks, thus zeroing the differential pressure that drives the coolant across the break. Also, the IRIS vessel has no penetrations at and below the core region, and it sits in an open cavity which extends above the core level. Suppression pools are located in the containment and they can double as gravity makeup. Decay heat is removed by four diverse (three independent) systems: eight steam generators, four natural circulation heat exchangers located outside the containment, surface (air and water) containment cooling.

  Loss of flow accidents (LOFAs) have no significant consequences because of the pumps characteristics and redundancy, as well as the substantial degree of natural circulation. Steam generator tube rupture accidents have lower probability and more benign consequences since the tubes are in compression (primary coolant outside) and are designed for zero internal pressure.

  The conclusion is that IRIS is a water reactor design where primary coolant related accidents are of no major concern, and of the eight Class IV accidents typically considered for LWRs, only one (refueling accident) remains as a Class IV accident. Three are eliminated by design and the remaining four can be reclassified at Class III or lower.

- An IRIS model using the RELAP5 code was completed to perform initial plant safety assessment and was verified in a preliminary steady state and transient qualification. All the relevant transient events and accidents typically reported in a Safety Analysis Report were assessed, including steam system piping failure, feed system piping failure, loss of offsite power, turbine trip, loss of flow, locked rotor, reactivity anomalies, steam generator tube failure, small break LOCA, and anticipated transients without scram.

In accordance with standard procedures, the system code (RELAP) was coupled with subchannel and neutronic analysis codes when required by the specific event considered. CFD analyses of selected portions of the pressure vessel have been performed to verify mixing phenomena in the IRIS system. For the analyses of small break LOCA, the strong coupling between vessel and containment during most of the event duration has required development of new approaches for the system analyses. While different solutions have been explored, a thermal-hydraulic coupling of RELAP (for reactor coolant system analysis) and GOTHIC (for containment analyses) was identified as the most promising approach and used in the analyses.

Models and results of the analyses have been collected in a preliminary plant safety assessment document which has been submitted to the NRC as part of the IRIS pre-application review. Probabilistic Safety Assessment (PSA) analyses have been initiated, with a preliminary assessment of event and fault trees, and the performance of a level 1 preliminary assessment. Indications are that the IRIS core damage function is orders of magnitude lower than current and advanced LWRs.

- Substantial work has been completed to support the IRIS goal of a 48-month interval between maintenance shutdowns. This, coupled with the extended core lifetime between refuelings, will yield very high capacity factors and significantly reduce the operating and maintenance (O&M) costs. A previous effort was performed by MIT to investigate the feasibility of extending the maintenance interval in a commercial PWR from 18 to 48 months. MIT identified 3,743 maintenance items for the 18-month cycle, 1,206 to be performed on-line and 2,537 off-line during the scheduled outage. The MIT study showed that most of the 2,537 off-line items could be deferred to 48 months or be performed on-line. Only 54 items in various categories (e.g., relief valves, motor operated valves, pumps, etc.) remained outstanding, as they still required an 18-month maintenance interval.

  Building on this study, the unresolved items were examined for their applicability to IRIS (e.g., pump oil lubrication obviously does not apply to the reactor coolant lubricated internal spool pumps). Only seven items in five categories were finally identified as still outstanding impediments to a 48-month maintenance interval in IRIS. They have been addressed and either solved or various plans for solutions have been designed. An additional category was identified as items which could be tested online, but would require a reduced power level for the test.

- As part of the Early Site Permit (ESP) programs, two potential arrangements of multiple IRIS modules were identified: one (1,000 MWe total) consisting of three modules “in a string” with staggered construction start and one (1,340 MWe total) consisting of two twin units where each unit has two modules sharing most of the systems.

- A market analysis and preliminary top-down cost estimate was performed, confirming the competitive attractiveness of IRIS, both in developed and emerging countries. The total cost of electricity was on the order of $0.03/kWh, with a capital cost around 1200 $/kWe.

- A pre-application licensing process was initiated with NRC on October 2002. The focus was two-fold: obtain review and concurrence of the IRIS testing program and of the approach to eliminate the need for off-site emergency response planning. As a first step, a documentation of the IRIS design and its safety analyses has been completed and provided to NRC. Regarding the first objective (testing program review), various activities are underway, which include preparation of PIRT (Phenomena Identification and Ranking Table) identification of required tests, specification of parameters, assessment of similitude analyses, preparation of test plans, and identification of test facilities.

Elimination of off-site emergency response is a goal stated by DOE for Generation IV reactors. It is believed that IRIS can satisfy such goal by supplementing the vastly enhanced defense in depth due to the safety-by-design with a focused risk informed regulation approach. Review by NRC of the IRIS project approach will be the second and final objective of the pre-application.

IRIS development does not end with the conclusion of the NERI three-year program. The IRIS consortium is proceeding with detailed design and analyses, focused on the NRC licensing process. The project schedule calls for initiating the formal Design Certification by the end of 2005, with attainment of Design Certification by 2008-2010 and deployment of the IRIS first-of-a-kind by 2012-2015.

1. INTRODUCTION

IRIS (International Reactor Innovative and Secure) is a next generation, 1000 MWt (~ 335 MWe) integral PWR which has been under development since late 1999. The Secure Transportable Autonomous Light Water Reactor – STAR-LW, was one of the proposals selected in the first year of the NERI program and its funding under the NERI grant terminated on August 2002. Termination of funding did not mean at all termination of the development of this new project. The IRIS reactor (as STAR-LW was renamed in October 1999) is in fact currently being developed by an international consortium of 20 plus organizations from nine countries, led by Westinghouse Electric Co. This very fact is in itself a testimonial to the success of the NERI program. NERI has provided the seed money and the catalyst for a new reactor design to take form, and thus IRIS represents the embodiment of the NERI program ultimate objective.

The fact that IRIS has evolved into an ongoing international cooperation project makes this program quite different from the other 1999-2002 NERI projects. Such difference had its impact on the timing and structure of this report. First of all, this is not a final report, since the work keeps progressing at an increasing pace, which is also the very reason why this report is being written almost one year after the official end of the NERI grant. In fact, in 2003 the efforts of the consortium have been almost exclusively focused on its pre-application licensing with the NRC. Consequently, this report documents the effort performed by the IRIS team over a four-year period, rather than the three-year NERI period. Next, the work performed and reported here goes well beyond what was originally envisioned in the NERI proposal, in depth and breadth, as well as in approach and philosophy.

The dramatic increase in the depth and breadth of scope was made possible by the contributions of the consortium members through self-funded studies as well as the transfer of related existing technology, including experimental data. It is estimated that the value of the self-funded consortium contributions is in excess of 90% of the total IRIS effort (and this does not include the technology transfer), thus providing a tremendous leverage to the DOE grant.

Regarding the approach and philosophy, the original NERI proposal essentially envisioned a three-year scoping and feasibility study aimed at defining the characteristics of an integral reactor conceptual design. This design would feature advanced cores requiring new technology developments and was to eventually evolve into a reactor plant deployable in the 2020-2030 period.

IRIS, on the other hand, following a first year of trade-off studies, focused on pursuing a commercial plant deployable in the 2012-2015 time frame. While maintaining and emphasizing the new engineering of the integral configuration for enhanced safety, simplicity and economics, the IRIS design relies as much as possible on the proven LWR technology and on the Westinghouse advanced passive plant designs AP600 and AP1000 to meet the target deployment dates.

The second year was therefore devoted to establish the commercial plant characteristics and to complete its conceptual design, while the third year concentrated on developing the IRIS preliminary design. The focus was on those aspects of the integral design which required (or allowed) improved, new engineering solutions. Those design aspects which could safely and conveniently rely upon the existing AP passive design solutions, were essentially “left to be filled in later”. Thus, the IRIS preliminary design, has adopted what we have called a “leopard skin” approach, where we first concentrate on the “spots” which require new engineering and eventually we enlarge the spots to fill in the whole skin. This approach allowed the project to proceed to a point where it could engage the NRC on a focused pre-application licensing process, which was initiated in late 2002.

Thus, in the current fourth year, the consortium efforts have been mostly devoted to support the design, analyses and preparation of the documentation necessary to proceed with the pre-application process.

Documentation to date of the IRIS project is vast and multi-faceted. Starting in 2000 and progressively increasing to date, a large number of open literature papers and articles have been published. Initially they were collectively authored by Westinghouse and the other participants, but as the work progressed, all IRIS consortium members published independently the results of their efforts. It should be noted that even though the consortium effort was internally funded, its results were for the most part made public, with only a few aspects kept as commercially confidential. A list of the IRIS papers and articles is reported in Appendix A. Two annual NERI reports documenting in great detail the work performed in the first[1] and second[2] year have been previously published. A description of the IRIS plant has been prepared and input to the latest edition of the IAEA TECDOC on advanced light water reactors.[3]

To make this final NERI report manageable and useful to the reader, previously published information has been reorganized in a structured way to present an overall picture of the IRIS design, starting with the early evolution and then focusing on the various design aspects. While the progress in the design is tracked as necessary, emphasis has been placed on presenting the various facets of the design as they currently are as of Fall 2003. Such presentations will be exhaustive, but brief and the reader is referred for more details to the previous two yearly reports or to the open literature publications, as appropriate.

Since this document will be comprised of many sections, a roadmap to them is provided next, as the project highlights and major accomplishments are summarized.

2. DESIGN HIGHLIGHTS AND MAJOR ACCOMPLISHMENTS

Following are the most salient aspects of the IRIS effort and the section in this report where they are presented in more detail.

- IRIS is developed by an international consortium of 20 plus organizations from nine countries and includes among its members industry, national laboratories, academia and power producers (see Section 3).

- Tradeoff studies were conducted in the areas of core neutronics and thermal-hydraulics to achieve enhanced safety and proliferation resistance. In particular full natural circulation (safety) and very long life cores (proliferation resistance) were investigated (see Section 4).

- IRIS is designed to be capable of accepting different cores. The core of the first IRIS module will have a fuel such to present no licensing issues. Thus, it has a fuel enrichment of 4.95%, i.e., less than the current 5% limit and a burnup within the current limit. On a straight burn life of four years a ~ 38,000 MWd/tU average discharge burnup is achieved. The burnup is increased to ~ 50,000 MWd/tU when moving to a half-core shuffle every 3.5 years and it can increase to the near term target of ~ 60,000 MWd/tU, with a third of the core shuffled every 2.5 years. The eventual choice will depend on utilities preferences (see Section 5.1). Future IRIS modules, depending on commercial considerations, could feature a fuel similar to the "first core" or a more advanced fuel. A preliminary design has been completed of cores with a 8-10 year straight burn life using higher (~ 8%) UO_2 enriched or MOX (~ 10% fissile) fuel in an open lattice (see Section 5.2). Finally, a very tight lattice core with exotic fuel shapes and over 15% enrichment, promises excellent performance, with a straight burn lifetime well in excess of ten years and average discharge burnups of the order of 120,000 MWd/tU (see Section 5.3).

- The primary system integral configuration has been defined and the preliminary design of the major components is proceeding. They include:

  - The reactor vessel and internals (see Section 6.1)
  - The steam generators (see Section 6.2)
  - The primary coolant pumps (see Section 6.3)
  - The pressurizer (see Section 6.4)
  - The neutron reflector (see Section 6.5)
  - The control rod drive mechanisms (see Section 6.6)

  The internal control rod drive mechanisms (CRDMs) have been a late addition. The IRIS position has always been that the integral reactor configuration is ideal for accommodating internal CRDMs, which have two very significant advantages: elimination of the vessel head penetrations and elimination of the rod ejection accident, but one disadvantage: not yet proven technology. In the first three years of the project, IRIS maintained the traditional external CRDMs as the reference design, with the internal ones as a backup until their feasibility were demonstrated. Recently, it has been decided to aggressively pursue the internal CRDMs as the reference design, spurred by the Davis-Besse incident, which has cost the utility (FENOC – First Energy Nuclear Operating Co.) about $500M from February 2002 to

June 2003, with a total cost before plant restart speculated to be liable to run close to \$1B. With many other reactors having experienced vessel penetrations problems prior to Davis-Besse and head replacement program actively underway among the utilities at the cost of tens of million of dollars per replacement, pursuing a design which inherently eliminates the problem root cause (which is impossible for loop PWRs) was an obvious choice.

As discussed in Section 6.6 two alternatives exist for the internal CRDMs: electromagnetic and hydraulic drives. A third alternative which also includes a new type of control rods, called "liquid control rods", where reactivity is controlled by the movement of a liquid absorber in a manometer type device was reported in the IRIS Year 2 Annual Report.

Feasibility studies on the hydraulically driven system, including proof-of-principle experiments have been performed by POLIMI (see Section 7). At the time of this writing, the electromagnetic drive is the preferred configuration, but active design effort has not yet been initiated.

- The integral configuration yields a large annular downcomer below the steam generators, separating the core from the reactor vessel. A very favorable consequence of this configuration is that the vessel fluence is decreased by several orders of magnitude. The typical PWR lifetime fast neutron fluence of 10^{19} n/cm^2 is reduced in IRIS to less than 10^{14} n/cm^2. There are multifold beneficial implications from economic, environmental and workers protection viewpoints:

  – The reactor vessel has practically no neutron damage and does not need replacement or annealing. From the standpoint of radiation damage, the reactor vessel life has no limitation.

  – Implementation of the vessel surveillance program and coupons sampling are not necessary.

  – The outer surface of the reactor vessel is "radiation-cold" and thus there will be essentially no exposure to crew working in the containment.

  – The reactor vessel can act as a sarcophagus for the reactor internals, (i.e., the irradiated internals, minus the fuel, can be left inside the vessel), thus greatly simplifying decommissioning and transportation.

  – The biological shield can be substantially reduced.

  – The effect of the water downcomer can be further increased by additional orders of magnitude when inserting shielding plates. The effects and characteristic of additional shielding have been studied (see Section 8). Economic considerations will eventually determine if additional shielding is warranted, given the fact that the water downcomer already provides significant radiation attenuation. Resolution is left for the final design, upon customers input.

- The IRIS design features an optimized maintenance approach, such that the interval between scheduled maintenance shutdowns is no shorter than 48 months. This, coupled with the extended fuel cycle, provides substantial savings in O&M costs and increase in capacity factor (see Section 9).

- A key distinguishing characteristic of IRIS is its approach to safety. IRIS safety is based on three tiers, anchored by its unique "safety-by-design". The safety-by-design represents the first tier and consists of designing the reactor such that accidents cannot possibly occur, or if they do occur, their consequences and/or probability of occurrence are intrinsically lessened by design, without the intervention of any engineered system, either active or passive (see Section 10.1). The second tier is represented by the passive safety systems to protect against and/or cope with the consequences of accidents unaffected or only partially affected by the safety-by-design. These systems (see Section 10.2) are similar to the LWR passive designs (mostly the PWR AP600/AP1000, but also some BWR type passive systems), but thanks to the safety-by-design, they are simpler and less in number. The third tier are active systems, as in current LWRs. However, there is a fundamental difference: in IRIS they are adopted to respond to normal and abnormal operating conditions and positively affect the PRA evaluation of core damage frequency, but they do not perform any safety function, as they do instead in current LWRs.

  Preliminary transient and accident analyses have been performed to quantitatively substantiate the IRIS safety approach. Focus has been on those accident sequences where IRIS behaves differently from loop PWRs (see Section 10.3). The RELAP code has been used for the transient analyses (see Section 10.4). These analyses have confirmed that the IRIS safety is superb. The bottom line is that of the eight Class IV accidents postulated for loop PWRs, only one remains in IRIS; three are eliminated outright and the remaining four are downgraded to a lower class.

- The IRIS containment design is an integral part of the safety-by-design. It is a small, spherical, high-design pressure steel vessel. In addition to the usual containment functions, it has a critical role in limiting the break flow during a small-medium LOCA (large LOCAs are eliminated by the integral configuration). The IRIS core remains covered throughout a LOCA accident, without the need for any emergency safety injection. Thus, IRIS does not have a high pressure safety injection system, offering enhanced safety and simplicity.

- The integral configuration provides both opportunities and challenges for plant control. Work has been initiated to outline an advanced state-of-the-art approach (see Section 11).

- The rapid development of IRIS has prompted the three utilities (Dominion, Entergy, Exelon) involved in the Early Site Permit (ESP) program to include IRIS in the group of reactor designs considered to provide a characteristics envelope for their designated sites. Two IRIS plant configurations have been developed to satisfy the ESP requirement of at least 1000 MWe installed. The first is a three module configuration where each module is independent; the other is a two twin modules unit (1340 MWe total) where each twin unit maximizes shared components (control room, fuel handling, radwaste, support systems, switchyard, etc.). The modular IRIS configuration allows to minimize construction time and financial exposure and also provides the utility with generating capacity and cash flow while subsequent units are in progress (see Section 12).

- A top-down evaluation of the projected IRIS economics has been performed, using the same methodology employed to assess the economics of PBMR and AP1000. Indications are that the IRIS economic goals of ~ $1200/KWe capital cost and ~ 3¢/kWhr cost of electricity are possible, making IRIS competitive in the entire energy production market (see Section 13).

- After exploratory meetings with NRC Commissioners, staff and ACRS as reported in the 2nd year report, IRIS officially initiated the pre-application licensing on October 2002. The pre-application licensing, to last until mid-2005, will focus on two items, one of which was selected because of its impact on cost/schedule and the other because of its novelty and complexity. The first is NRC review and eventual agreement with the IRIS test program necessary to obtain design certification. The second is NRC review and feedback on the IRIS approach to adopt a "focused" risk informed regulation, aimed at enhancing the IRIS licensing objectives, such as to eliminate the requirement for off-site emergency response planning outside the exclusion zone. The underlying rationale is that the IRIS safety-by-design presents such an improved deterministic defense in depth that a PRA based risk-informed approach could demonstrate that the DOE goal for Generation IV reactors of no off-site emergency response can be attained by IRIS (see Section 13).

3. THE IRIS INTERNATIONAL APPROACH AND CONSORTIUM

Almost immediately after the NERI award for the STAR-LW effort, Westinghouse and its three university partners (MIT, University of California at Berkeley, and the Polytechnic of Milan, Italy) agreed that the focus would be on developing a commercially viable concept and thus avoid its becoming just one more paper reactor like so many of its predecessors. It was evident that the era of a single company, or even a single nation, developing and deploying a nuclear plant had past. Also, it was apparent that many utilities, as well as developing nations, are interested in capping their capital investment in a power plant project to only a few hundred million dollars, thus driving them to concentrate on smaller capacity additions. Larger plants, however, have economy of scale and therefore a new dimension has to appear for smaller plants to become more economical and true market competitors.

The unique potential economic advantages of small modular reactors were investigated[4] in the 90's by the SIR, another water cooled integral reactor, in several aspects an IRIS predecessor. More recently, smaller, modular gas cooled reactors had been proposed, the PBMR (Pebble Bed Modular Reactor)[5] and the GT-MHR (Gas Turbine-Modular Helium Reactor).[6] For the PBMR, Exelon had made a strong case of the inherent advantage of small plants in introducing new power to the grid in limited increments, thus finely tailoring supply and demand and limiting the utilities' financial exposure. IRIS of course shared all those considerations, and it also emphasized that, in addition to being simpler to construct and operate, these smaller plants had to be fabricated in series. It was readily apparent that to fabricate and deploy an economically large enough number of multiple, identical modules, the market had to be one global, international arena.

Once it was established that this new reactor was to be deployed world-wide, it followed that to be readily accepted internationally, it had to be developed internationally, i.e., it had to address international requirements, needs and even cultures. Hence the change of name from STAR-LW to IRIS to emphasize with the first letter (International) of its acronym that, from the very beginning, IRIS was going to be designed and subsequently fabricated, deployed and serviced by an international partnership, where all team members were stakeholders in the project.

This approach immediately found a positive resonance, as the IRIS team kept growing in its first three years from the initial four members and two countries to the present 20 plus members from nine countries (see Table 1 and Figure 1). The original team was joined by other reactor designers and component manufacturers, fuel vendors, architect engineers, power producers, universities, and laboratories. Table 1 provides a summary of the IRIS team partnership with the areas of responsibility of each team member. Associate members are U.S. universities and laboratories currently working on DOE funded Nuclear Energy Research Initiative (NERI) projects, which, while of general interest, use IRIS as the example application of the technology being investigated.

The IRIS consortium members are self-funded and provide to the project both design effort and previous know-how. Currently, approximately 100 people across the IRIS consortium are working on the IRIS design.

Table 1 Member Organizations of the IRIS Consortium

INDUSTRY		
Westinghouse	USA	Overall coordination, core design, licensing
BNFL	UK	Fuel and fuel cycle
Ansaldo Energia	Italy	Steam generators design
Ansaldo Camozzi	Italy	Steam generators, CRDMs fabrication
ENSA	Spain	Pressure vessel and internals
NUCLEP	Brazil	Containment, pressurizer
Bechtel	USA	BOP, AE
OKBM	Russia	Testing, desalination
LABORATORIES		
ORNL	USA	I&C, PRA, shielding, pressurizer, core analyses
CNEN	Brazil	Pressurizer design, transient and safety analyses, desalination
ININ	Mexico	PRA support
UNIVERSITIES		
Polytechnic of Milan	Italy	Safety analyses, shielding, thermal hydraulics, steam generators design, internal CRDMs, desalination
MIT	USA	Advanced cores, maintenance
Tokyo Inst. of Technology	Japan	Advanced cores, PRA
University of Zagreb	Croatia	Neutronics, safety analyses
University of Pisa	Italy	Containment analyses
Polytechnic of Turin	Italy	Human factors, reliability availability maintainability support
University of Rome	Italy	Radwaste system, occupational doses
POWER PRODUCERS		
TVA	USA	Maintenance, utility perspective
Eletronuclear	Brazil	Developing country utility perspective
ASSOCIATED US UNIVERSITIES (NERI PROGRAMS)		
Univ. of California Berkeley	USA	Neutronics, advanced cores
Univ. of Tennessee	USA	Modularization, I&C
Ohio State	USA	In-core power monitor, advanced diagnostics
Iowa State (& Ames Lab)	USA	On-line monitoring
Univ. of Michigan (& Sandia Labs)	USA	Monitoring and control

The contribution of the universities to the IRIS program cannot be emphasized enough. Innovative design solutions have been proposed and developed by universities and IRIS is perhaps the first and only commercial reactor project where academia and industry are in a partnership equally co-responsible for the design. The partnership with universities (and laboratories) has also a potentially very important long-term effect, in making IRIS a “living and contemporary” design. In fact, once the IRIS preliminary

Figure 1 IRIS Logo

design is completed, its implementation becomes essentially the responsibility of the industrial partners, while the universities and laboratories will shift to work on future, even more improved designs to incorporate the most recent technological advancements. As they are readied, industry can then implement them in a new series of IRIS modules. A key reason that this can conceivably be done and accepted by the market is that the size of an IRIS module is only about one-third to one-fourth of today's large light water reactors (LWRs) and thus the financial exposure is much more limited.

As of Fall 2003, 97 students at all levels (BS, MS, PhD) have or are contributing to the IRIS project at ten universities in four countries (see Table 2).

A total of 59 graduate theses (45 MS, 14 PhD) have been completed or are in progress (see Appendix B). 41 students have so far graduated with a thesis on IRIS. Twelve students have been or currently are interns at the Westinghouse Science and Technology Department.

The IRIS project is most proud of its record of providing so many students around the globe with the opportunity of working on a cutting edge technology reactor design and learning design skills "on the job". On the other hand, the project is most grateful to these students, because they have given fundamental contributions enabling IRIS to be cutting edge technology.

Table 2 Students Contributing to IRIS Design

University	Undergraduate	Graduate	Doctorate
Polytechnic of Milan, Italy	1	20	4
MIT, USA	1	4	1
Univ. California Berkeley, USA	-	2	-
University of Pisa, Italy	26	4	1
Tokyo Institute of Technology, Japan	-	3	4
University of Tennessee, USA	1	4	-
Ohio State University, USA	-	4	1
University of Michigan, USA	6	2 (planned)	-
University of Zagreb, Croatia	3	1	3
Polytechnic of Turin, Italy	-	1	-
	38	45	14

4. TRADE-OFF STUDIES

The first year of the project, till the end of 2000, was dedicated to performing various trade-off studies, which have been reported in detail in the first year annual report.[1] Essentially these studies enabled the transition from the highly developmental reactor concept envisioned in the STAR-LW proposal (very small ~ 50 MWe size, 15 years straight burn core, hardened spectrum core with exotic fuel geometries, full natural circulation with limited coolant boiling) to a more realistic design (300 MWt, 8 to 10 years straight burn core, thermal spectrum with standard fuel geometry, forced circulation). Still, at the beginning of 2001, the project shifted again, for economic and schedular considerations, to the present design of 1000 MWt (~ 335 MWe), four years or less core life, fuel enrichment and burnup within currently licensed limits.

Reported below is a summary of the trade-off studies; for additional details the reader is referred to Reference 1.

In conducting the IRIS trade-off studies, the NERI solicitation four requirements for Generation IV designs had to be satisfied:

1. Proliferation resistance. This was quantitatively translated in minimizing access to the fuel by the host country through a long life straight burn core without shuffling or refueling.

2. Improved economics. All possible solutions should result in capital or operating costs improvement.

3. Enhanced safety. IRIS approach is "safety by design" where by design most accidents either cannot occur or their consequences/probabilities are lessened.

4. Waste reduction. This also included approaches to simplify decommissioning.

As the IRIS development progressed, a fifth requirement came to the forefront. As the prospects for a nuclear revival brightened dramatically in 2000 with utilities actually thinking and talking about new construction, it became evident that the modular IRIS, which relies on proven LWR technology while offering substantial improvements, was a candidate for medium term deployment. Thus, the additional requirement was to adopt technical solutions which could be confidently deployed by 2010. More advanced solutions which require longer technology demonstration could be pursued for eventual implementation in subsequent plants.

To better understand the discussion of the tradeoff studies, however, a brief description of the IRIS design as it was developing in 2000 follows.

This design featured an integral vessel which houses the reactor core and support structures, core barrel, upper internals, control rod guides and drivelines, steam generators, pressurizer, heaters, and externally mounted canned motor reactor coolant pumps (see Figure 2). Such an arrangement eliminates separate steam generators and pressurizer, connecting pipes, and supports. Depending on the plant power rating, the vessel has a height of 18-22 m and an outside diameter of 4-6 m, a size which is within the state-of-the-art fabrication capabilities. The configuration shown in Figure 2 is for a

300 MWt design. Hot coolant rising from the reactor core to the top of the vessel is pumped into the steam generator annulus by six reactor coolant pumps. Axial location of the pumps depends on the trade-off between the deteriorated pump performance at high coolant temperature, and the desire to eliminate low vessel penetrations near the core. The top location shown in Figure 2 was the preferred position, which was confirmed for all subsequent designs.

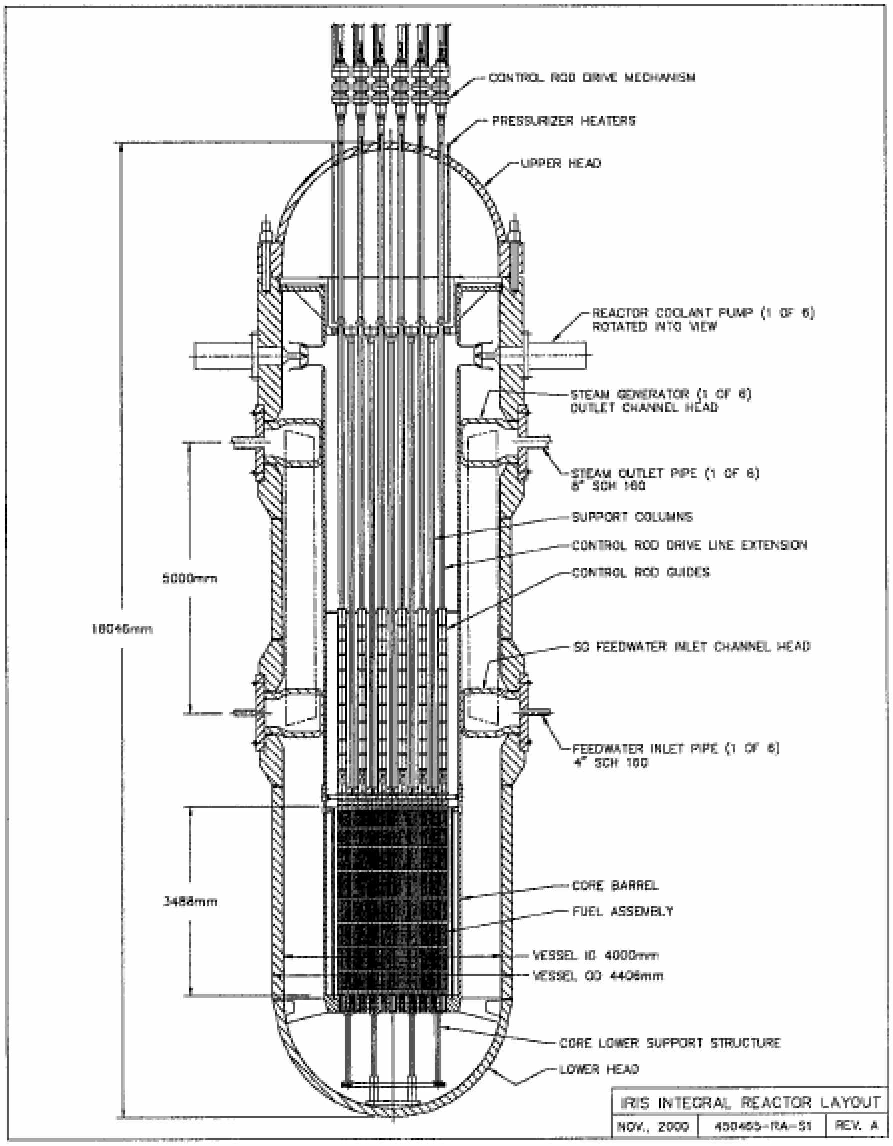


Figure 2 Layout of the IRIS Primary System (Year 2000 Design)

The following sections discuss the key trade-off studies which were conducted to arrive at the selection of core characteristics. Even though no reasonable alternative was excluded a priori, the various design parameters and choices are for the most part not at all independent, and thus only a limited set of conditions was found to satisfy the previously stated requirements.

4.1 CORE NEUTRONICS

Major factors determining the initial core neutronics design are fuel form and enrichment, fuel lattice, fuel cycle and cladding selection. They are discussed in the following.

4.1.1 Fuel Form Selection

Different fuel forms were initially considered, including enriched UO_2, mixed U-Pu oxide (MOX), metal fuel, carbide and nitride fuel, and dispersion fuels.

The maximum fissile content was set at 20% ^{235}U, according to the DOE specified upper limit to satisfy proliferation resistance considerations. While all the considered materials offered some unique advantages, satisfaction of the second (economics) and fifth (no major developments) requirements limited the choices to the proven UO_2 and MOX fuels. Both fuels as it will be seen in next Section 4.1.2 have acceptable, even though different, neutronics behavior. The cost of UO_2 fuel strongly depends on its level of enrichment, while in the case of MOX its fabrication and handling have a higher cost impact than the fissile content. Finally, the use of MOX is of interest to the IRIS international partners, but UO_2 is preferred by the US for proliferation resistance considerations. Therefore it was concluded that keeping both options open would be advisable, thus the decision to consider the UO_2 core as the reference, but to also implement a MOX core as an alternate design. Consequently, a characteristic feature of the IRIS design is the capability of operating with either a UO_2 or MOX core. This interchangeability can be accomplished in IRIS because of its unique characteristic of long life straight burn core with no shuffling.

IRIS is envisioned not to be a static design, but to evolve with advances in technology and thus to be able to later accept advanced solutions. This projected evolution is facilitated by two IRIS features: a) its modular and simplified design; b) the participation to the IRIS team by universities and laboratories who will keep working on advanced solutions, while the industrial members of the team concentrate on the deployment of the "first" IRIS. Therefore, the use of advanced fuels (thorium, cermet, dispersion) will be addressed in future studies.

4.1.2 Fuel Lattice Selection

For light water cooled reactors, the fuel lattice is commonly represented by the fuel-to-moderator ratio, expressed either by the ratio of heavy metal (U+Pu) to hydrogen atoms, or by the p/d ratio, where p represents the lattice pitch and d represents the cladding outer diameter. Generally, a tight lattice (small p/d combined with triangular/hexagonal lattice) leads to reduced neutron moderation, and increased fuel conversion due to a hardened (epithermal) neutron spectrum. Hence, initial fuel reactivity is lower, but the reactivity drop with depletion is slower. On the other hand, an open lattice (large p/d, allowing either square or triangular/hexagonal lattice) leads to better neutron utilization and higher initial reactivity, but also to faster reactivity drop with depletion.

This is of special importance for IRIS when the long core life is one of the design objectives. Figure 3 illustrates the neutronic behavior for a 10% enriched UO_2 fuel. Tight lattice reactivity (effective multiplication factor, k-eff) starts notably lower, but the slope of its reduction is smaller, and at some point it will break even with the open lattice curve. If this happens while k-eff > 1.0, tight lattice provides longer core life (in terms of the discharge burnup). However, for the particular case shown in Figure 3, k-eff for the open lattice remains higher in the k-eff > 1.0 region.

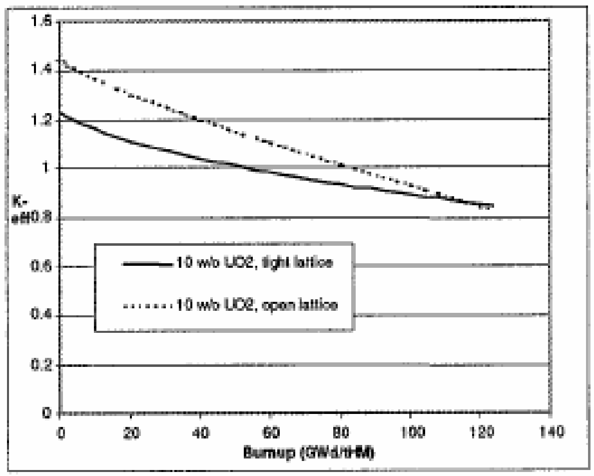


Figure 3 Core Reactivity for Tight and Open Lattice (10 w/o ^{235}U in UO_2)

Optimum selection depends on other parameters, primarily fissile enrichment and fuel form (UO_2 or MOX fuel). A series of lattice calculations were performed, obtaining the infinite multiplication factor k-eff corrected for the neutron leakage, and translating it into achievable discharge burnup for a straight burn. Results for two different enrichments are reported in Figures 4a and 4b, for UO_2 and MOX fuel respectively, where the discharge burnup is shown as a function of p/d in a square lattice. For fuel with approximately 10% fissile content, open lattice provides high discharge burnup in both cases (UO_2 and MOX). However, it is interesting to note that for higher fissile content, a tighter lattice provides an even higher discharge burnup for MOX fuel.

Even though the triangular lattice yields somewhat better neutronics performance and tighter assembly packing for small cores, a square lattice was chosen to take advantage of the large PWR experience base and available manufacturing capabilities.

An eight-year core life, achievable with a ~ 10% fissile content in an open lattice configuration, was chosen as the core design best satisfying both the proliferation resistance and economic requirements. This corresponds to an average discharge burnup in the 70-80,000 MWd/t range, which is a not-too-far extrapolation from the current data base. Higher performance, i.e., an extended core life up to 15 years without refueling might be achieved in a tight lattice with MOX fuel. However, the discharge burnup, of the order of 140,000 MWd/t, is more than double the current oxide fuel technology. Again, this will be the subject of future studies examining advanced fuel forms.

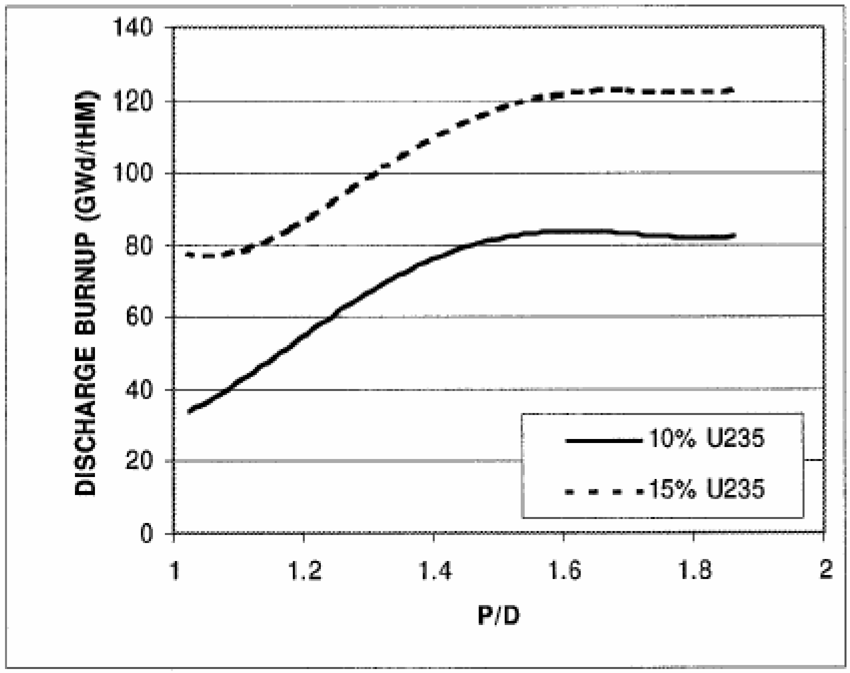


Figure 4a Discharge Burnup as a Function of p/d for UO_2 Fuel

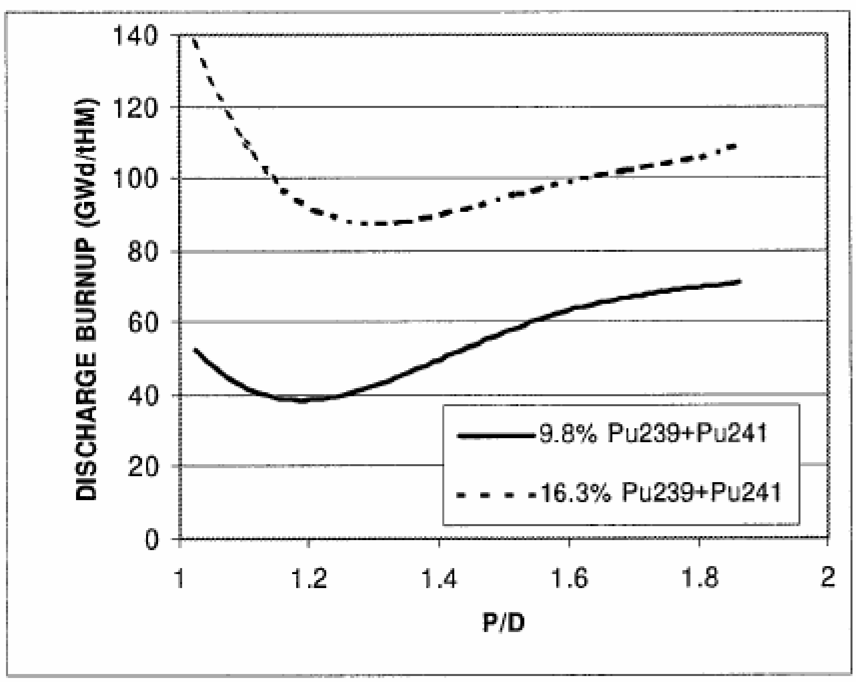


Figure 4b Discharge Burnup as a Function of p/d for MOX Fuel

4.1.3 Fuel Cycle Selection

A long-life core with no shuffling or refueling severely limits access to the fuel during reactor operation and therefore positively addresses the proliferation resistance requirement. It also has a positive economic effect by increasing the potential plant capacity factor into the high nineties percentage. With long-life cores, the maintenance outage becomes the limiting downtime interval which determines the capacity factor, and therefore the effort to design IRIS to four-year scheduled maintenance intervals, which will be discussed in Section 9.

Thus, stretching the core life to long intervals like 15 years, besides being impractical from a technological standpoint as previously seen, is uneconomical because very little is gained in terms of capacity factor, which is dependent on the maintenance interval, while uneconomical, very low power density cores are necessary. Thus, an eight-year core life, as previously determined from neutronics considerations, appears to be also the near optimum choice from an economic point of view.

4.1.4 Cladding Selection

Two main considerations were present in the cladding selection process. Primarily, from the safety standpoint, cladding must guarantee fuel integrity for the design burnup limit. Secondarily, for neutron economy, the cladding reactivity penalty has to be acceptable. For the eight-year core design employing open lattice, advanced Zircaloy cladding provides a viable solution in both respects.

If extended core design (up to 15 years lifetime) is pursued in the future, the average fuel burnup would significantly exceed 100,000 MWd/t, in a tight lattice and hard spectrum, consequently high fast neutron fluence would result. In this case, stainless steel cladding will most probably be the preferred choice since it provides the required material properties, while at the same time its reactivity penalty becomes acceptable because of the hard spectrum.

The effect of lattice parameter p/d on the fast fluence and DPA (displacements per atom) is illustrated in Table 3.

Table 3 Fast (E>1MeV) Neutron Fluence and DPA (E>1MeV) in Cladding for 15-Year Core Life (MOX with 20 w/o Pu, 3 kW/ft, based on BOL Spectra)

p/d	Fast Fluence (10^{22} n/cm^2)	DPA
1.00	6.0	62
1.10	4.7	49
1.17	4.0	43
1.25	3.4	37
1.40	2.6	29
1.55	2.1	23

Monte Carlo simulations were performed for lattices with different p/d ratios employing MOX fuel with 20% fissile content. The neutron spectrum was obtained and folded together with DPA cross sections for stainless steel, to obtain estimates of the lifetime fast neutron fluence and DPA over fifteen years of operation at 95% availability, assuming 3 kW/ft linear power rating. The obtained fluence is significantly higher than in PWRs, but it remains below $1x10^{23}$ n/cm^2 and should therefore be acceptable, i.e., it is not a limiting factor. It should be noted that the accumulated DPA roughly doubles with transition from a thermal spectrum to a tight lattice and harder spectrum.

4.2 CORE THERMAL-HYDRAULICS

The focus of the trade-off studies to determine the core thermal-hydraulics was to examine to what extent the IRIS design should feature natural circulation. Most of the integral type reactors reported in the literature, like NILUS,[7] CAREM[8] and more recently IMR,[9] feature full natural circulation to preclude loss of flow accidents. If that should indeed be the objective of IRIS, attainment of full natural circulation can be enhanced by adopting a high reactor ΔT (which will decrease the coolant flow, hence the pressure drop) and/or allowing core boiling (which will increase the density differential head). The OSCAR (Optimization Simplified Code for Analysis of integral Reactor) code, developed by the Polytechnic of Milan, and successfully used in the design of NILUS, was adapted to these analyses. Primary system key parameters were calculated for three different core configurations having p/d ratios of 1.05, 1.10 and 1.45. Full, single phase natural circulation was imposed in all three cases; the core inlet temperature was 275°C and the outlet temperature 330°C, yielding a reactor ΔT double the current PWRs value. The results are reported in Table 4.

Table 4 System Configurations Yielding Full Natural Circulation

		p/d = 1.45	p/d = 1.10	p/d = 1.05
Reactor power	MWt	300		
ΔT core	°C	55		
Average linear power	Kw/m	10.56	10.05	10.53
Vessel diameter	m	4.0	3.8	3.7
Required vessel height	m	33	69	136
Vessel weight	ton	618	1260	2355
SG pressure losses	KPa	17.8	20.1	21.9
Core pressure losses	KPa	4.3	38.9	101.2

It is obvious that for tight lattice cores, single phase full natural circulation is highly impractical because of the required vessel height. However, even for the moderated open lattice core, the required vessel height would be in excess of 30 meters, which is obviously uneconomical.

Further analyses were conducted to evaluate the effect on natural circulation of allowing core boiling. Various p/d configurations and vapor qualities were considered; as expected, boiling did enhance natural circulation and in the case of the 1.05 p/d, the vessel height was reduced from 136 m (Table 4) to 25 m for a 10% vapor quality and to 12 m for a 40% quality. For p/d ratios higher than 1.1, the vessel height was no longer a critical parameter. However, thermal analyses indicated that the margin to DNB (Departure from Nucleate Boiling) was unacceptable even at low vapor qualities unless uneconomically low power densities were adopted. Also to be accounted for are the peaking factors in an unorificed open core, thus leading to higher vapor qualities in the hot channels.

Another parametric analysis was conducted varying the IRIS thermal power and it was found that reactor designs with full natural circulation would be quite viable and even preferable for powers of 150 MWt or less, debatable in the range of 150 to 400 MWt, and completely unrealistic for powers above 400 MWt. Thus, all analyses agreed that for IRIS a full natural circulation design was not feasible. However, the advantages of natural circulation can be exploited in a design combining partial natural circulation with low head pumps. This is the optimal solution for IRIS which unlike loop-type LWRs has a configuration (integral reactor, elevated steam generators, open core lattice in the moderated version) naturally lending itself to enhanced natural circulation. Thus the IRIS design will feature "aided natural circulation," i.e., the total reactor flow will be comprised of pumped plus natural circulation flow. The fraction of reactor flow provided by natural circulation can be varied by appropriate choices of design configuration and characteristics.

Finally, trade-off studies were conducted to determine the optimum reactor ΔT. Natural circulation considerations favor a lower inlet temperature, which however is detrimental from the point of view of secondary side steam pressure and efficiency, as can be seen in Figure 5. The core outlet temperature is practically dictated by the primary system pressure and core outlet quality, so it is not too dissimilar from current PWRs. The selected core inlet and outlet temperatures for IRIS were 292°C and 330°C, respectively. The value of the inlet temperature can be reassessed if transient analyses indicate that a higher degree of natural circulation is needed for safety reasons.

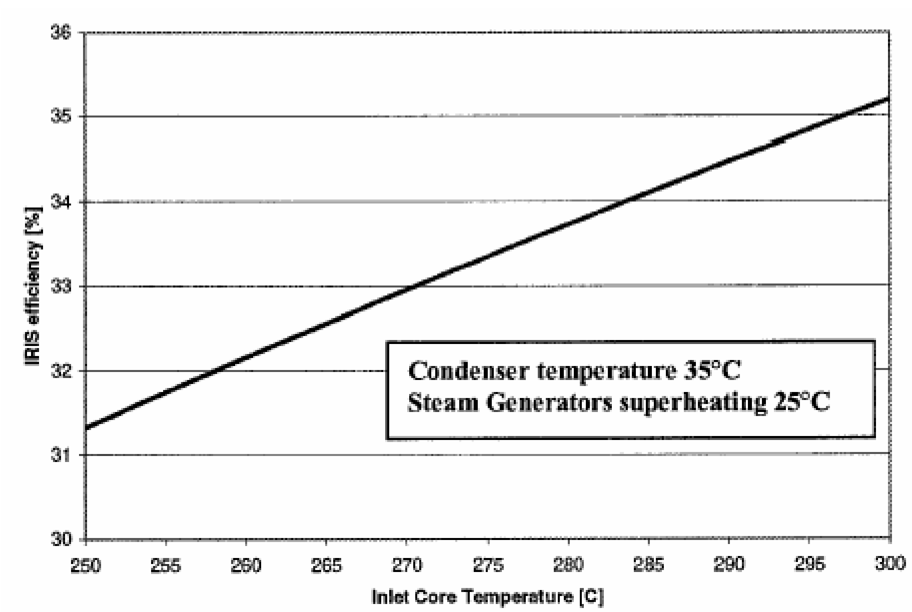


Figure 5 IRIS Efficiency as a Function of Inlet Temperature

5. CORE DESIGNS

As previously mentioned, IRIS is designed to accommodate various core designs depending on the projected time of deployment. For initial cores, today's licenseable fuel is adopted; for later cores, fuels with higher enrichments and higher burnup are considered; finally, developmental fuel and fuel assemblies capable of very high burnup in excess of 100,000 MWd/t and very long life in a straight burn mode are envisioned for deployment around 2030. These three designs are discussed in the following sections.

5.1 FIRST CORE DESIGN (2012-2015 DEPLOYABLE)

The fuel assemblies in the IRIS first core are similar to those of a loop type Westinghouse PWR design. Specifically, the IRIS fuel assembly design is similar to the Westinghouse 17x17 XL Robust Fuel Assembly design and the AP1000 fuel assembly design. An IRIS fuel assembly consists of 264 fuel rods with a standard 0.374" OD in a 17x17 square array. The central position is reserved for in-core instrumentation, and 24 positions have guide thimbles for the control rodlets. The core configuration consists of 89 fuel assemblies; this configuration has a relatively high fill-factor (i.e., it closely approximates a cylinder), to minimize the vessel diameter (see Figure 6). The IRIS 1000 MWt core has a low power density; the active fuel height is 14 ft. (4.267m) and the resulting average linear power density is about 75 percent of the AP600 value. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and increased overall plant capacity factors.

5.1.1 Straight Burn Option

In this option, the whole core is replaced at each reload. The fuel is UO_2, enriched to 4.95 w/o in ^{235}U, with lower enrichment (2.6 w/o ^{235}U) in the axial blankets and at the core periphery.

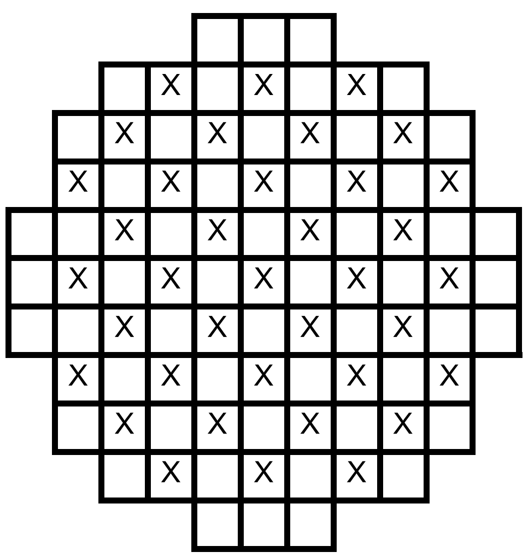


Figure 6 IRIS Core Configuration and a Typical Control Rod Pattern

Enriched boron, axially zoned, is used as an integral fuel burnable absorber, IFBA (IFBA is a thin layer of ZrB_2 coating the fuel pellets, and IFBA loading is expressed here in mg ^{10}B per cm of fuel rod). For fuel assemblies with IFBA, the 14 foot fuel stack is axially composed of 1 foot of enriched uncoated fuel above and below a 12 feet central region of enriched coated fuel. The lower half (6 feet) of the coated part of the fuel rod has 20% more ^{10}B than the upper coated part. The radial core configuration is shown in Figure 7.

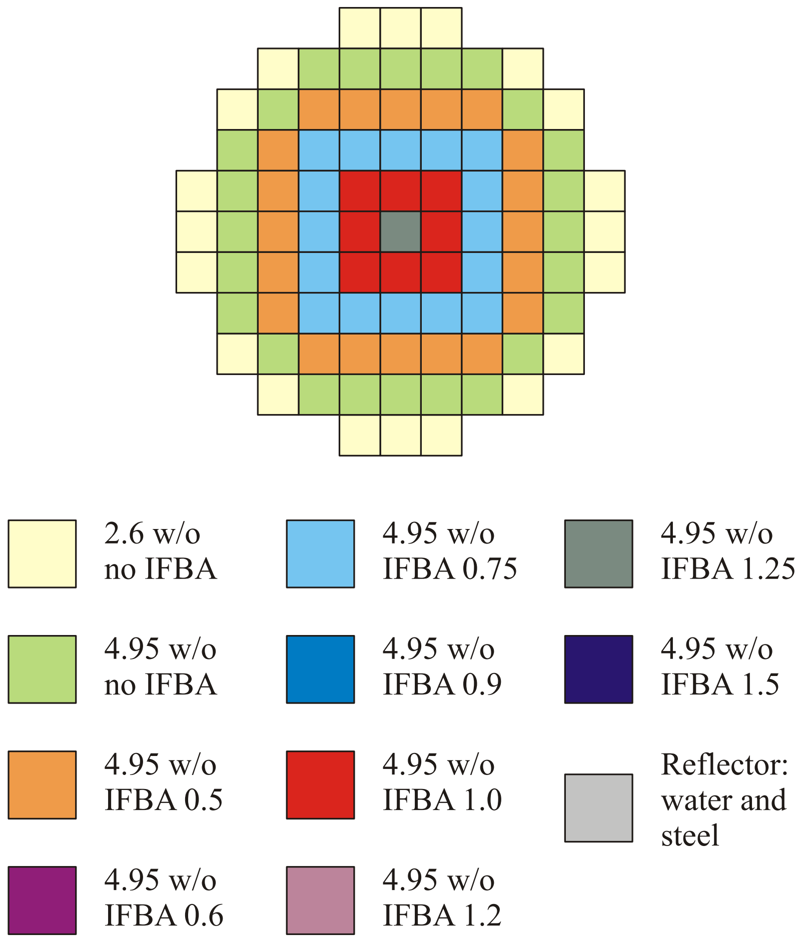


Figure 7 Radial Enrichment and IFBA Distributions in the Straight-burn IRIS Core

Reactivity control is accomplished through IFBAs, control rods, and the use of a limited amount of soluble boron in the reactor coolant. The reduced use of soluble boron makes the moderator temperature coefficient more negative, thus increasing inherent safety, and lessening boric acid induced corrosion concerns.

In addition to using IFBAs, erbium in form of Er_2O_3 mixed in the fuel is another standard Westinghouse integral burnable absorber. Figure 8 shows the estimated k_{eff} as a function of burnup, whereas two different linear densities of ^{10}B and one erbium concentration are considered. It may be observed that there is practically no reactivity

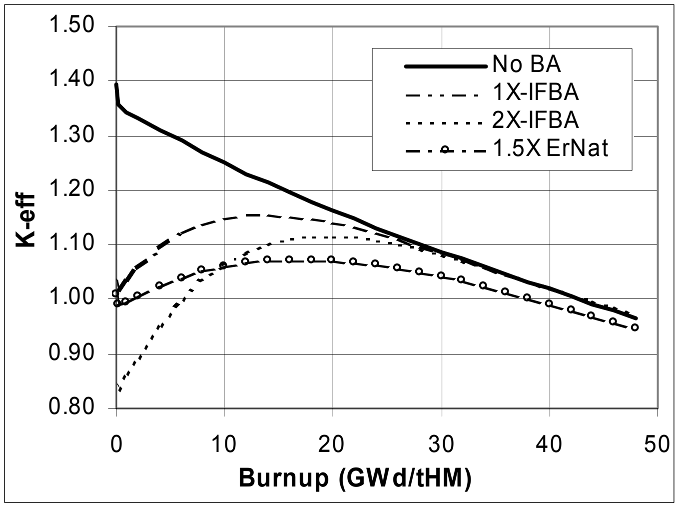


Figure 8 Effective Multiplication Factor as a Function of Burnup for IRIS UO_2 Fuel, 4.95% Enriched. IFBA and Erbium Burnable Absorber.

penalty for IFBA at burnup past 30,000 MWd/tU, and only a small penalty at ~25,000 MWd/tU. The depletion range in which the reactivity hold-down is significant is limited primarily to the first 10-15,000 MWd/tU, even though the considered IFBA loading (2 mg ^{10}B/cm) is several times higher than that used in present PWRs. Note that a large boron loading is acceptable in IRIS because its fuel is designed with a significantly increased (roughly doubled) fission gas plenum length compared to current PWRs, thus eliminating potential concerns with internal overpressure. The integral RV design permits this increase in the gas plenum length with practically no penalty, because the steam generators mainly determine the vessel height.

Erbium provides better reactivity control and much flatter k_{eff} profile then IFBA, but there is some non-trivial reactivity penalty. In fact, due to its lower cross section, ^{167}Er depletes slower than ^{10}B. However, the erbium isotope ^{166}Er that is present in natural Er is responsible for a relatively large residual reactivity penalty. Figure 9 illustrates both effects, i.e., a reasonably flat reactivity profile over the whole depletion, as well as the undesirable residual reactivity penalty (remaining difference in reactivity even for high fuel burnup).

The critical soluble boron concentration for an alternative core configuration utilizing erbium alone as burnable poison is shown in Figure 10. It should be noted that the boron concentration remains below 800 ppm throughout the cycle, however, the cycle length is reduced (by ~4,000 MWd/tU) as compared to cores utilizing IFBA alone. Therefore, an optimum burnable absorber design for an IRIS straight burn four-year cycle could combine erbium (extended reactivity suppression) with IFBA (no residual reactivity penalty). Several core configurations employing erbium in addition to IFBA have been devised that have the potential to satisfy all design requirements.

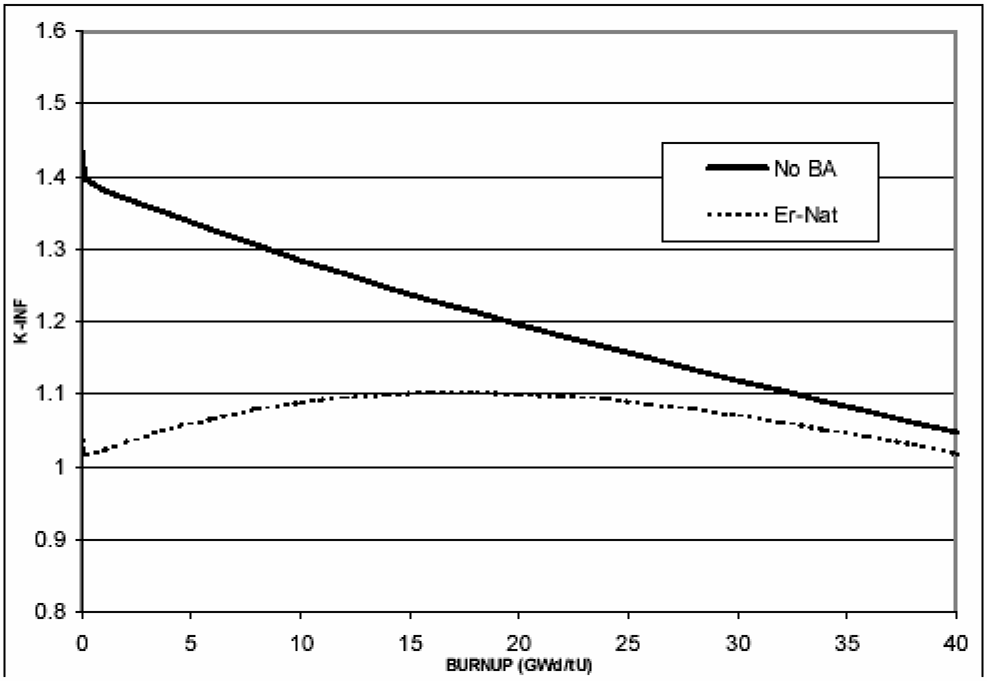


Figure 9 Infinite Multiplication Factor with and without Er Burnable Absorber

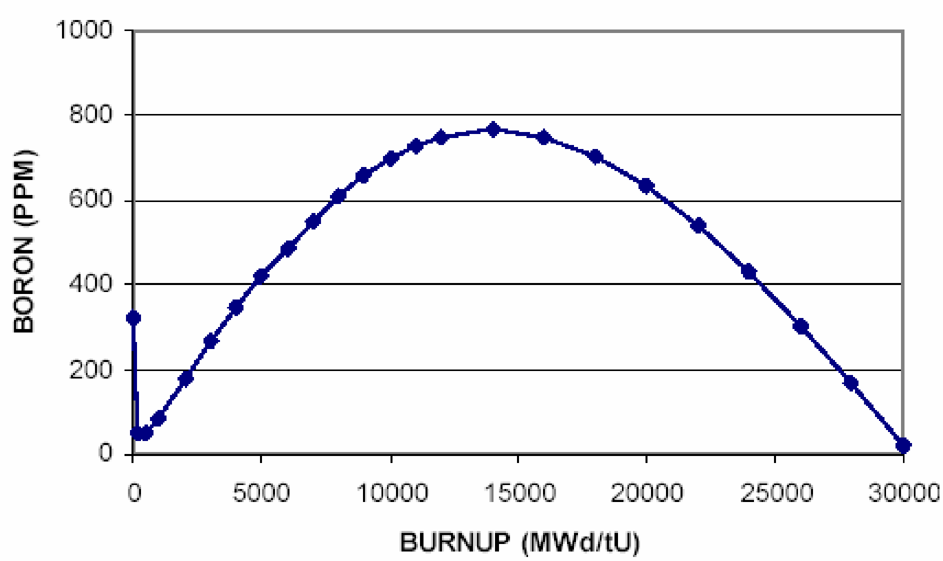


Figure 10 Critical Soluble Boron Concentration for a Representative Core Configuration Employing Erbium

5.1.2 Multi-Batch Reload Option

Previous analyses have shown that a straight burn 4-year cycle can be achieved without complications. However, a straight burn 48-month core has a relatively low discharge burnup (~ 40,000 MWd/tU for 4.95 w/o ^{235}U fuel, and even lower for 2.60 w/o ^{235}U fuel) and feedback from utilities indicated that a high burnup was preferable to a longer single core cycle length. Therefore designs featuring two-batch and three-batch cores, with partial refuelings were also developed (see Table 5).

Table 5 Refueling Options for IRIS First Core Design

	Emphasis on Proliferation Resistance	Reference Option	High Burnup Option (When Licenseable)
	Single Batch (Straight Burn)	Two-Batch (Partial Reload)	Three-Batch (Partial Reload)
First Core	69 FAs @ 4.95% 20 FAs @ 2.6%	44 FAs @ 4.95% 45 FAs @ 2.1%	52 FAs @ 4.95% 37 FAs @ 2.1%
Reload	same	40-44 FAs @ 4.95%	28-36 FAs @ 4.95%
Cycle Length (yrs)	4.0	3.0-3.5	2.5-3.0
Average discharge burnup (MWd/tU)	38-40,000	46-53,000	56-62,000
Lead rod burnup (MWd/t	<50,000	<62,000	<75,000

The current reference design is the two-batch core, having a cycle length in excess of 3 years and a lead rod burnup less than 62,000 MWd/tU, which is consistent with the currently licenseable limit. Once that limit is raised to 75,000 MWd/tU, as currently envisioned, IRIS can easily keep pace by going to a three-batch core.

The equilibrium cycle in the two-batch reloading strategy is approximated in the first cycle by a split-feed core configuration, where one-half of the fuel assemblies (shown in gray in Figure 11) have reduced enrichment (2.1 w/o), emulating once-burnt fuel. Also shown in this figure is the erbia loading distribution.

The first core design has therefore been selected in order to present no licensing problems and to have the flexibility to satisfy utility and/or proliferation resistance requirements. In addition, as already mentioned, the IRIS design is such to accept interchangeable cores, offering the option to later use the advanced cores discussed in the next sections.

1 2.10 w/o No BA	2 4.95 w/o 1.8%	3 2.10 w/o No BA	4 4.95 w/o 1.8%	5 4.95 w/o 1.8%	6 2.10 w/o No BA
7 4.95 w/o 1.8%	8 2.10 w/o No BA	9 4.95 w/o 1.8%	10 2.10 w/o No BA	11 4.95 w/o 1.8%	12 2.10 w/o No BA
13 2.10 w/o No BA	14 4.95 w/o 1.8/1.5%	15 2.10 w/o No BA	16 4.95 w/o 1.8/1.5%	17 4.95 w/o No BA	
18 4.95 w/o 1.8%	19 2.10 w/o No BA	20 4.95 w/o 1.8%	21 2.10 w/o No BA	22 2.10 w/o No BA	
23 4.95 w/o 1.8%	24 4.95 w/o 1.8%	25 4.95 w/o No BA	26 2.10 w/o No BA		
27 2.10 w/o No BA	28 2.10 w/o No BA			 Enrichment BA (erbia w/o)	FA# 4.95 w/o 1.8%

Figure 11 Radial Enrichment and Erbia Distributions in the Two-Batch IRIS Core

5.2 LONGER LIFE CORES (2020 DEPLOYABLE)

The IRIS core interchangeability is made possible by the adoption of the variable moderation approach. As seen in Section 5.1.1, a 4.95% UO_2 fuel using current, licensed fabrication technology provides a four-year core lifetime. Longer lifetimes are achievable with higher enrichment, which is currently not licensed, but it could conceivably be in the next 5-10 years. As seen in Section 4.1.2, an eight-year core lifetime was achievable with UO_2 or MOX fuel in the 8-10% fissile range.

Higher uranium enrichment, and even more so MOX fuel, require increasing the moderating ratio V_m/V_f to retain good fuel utilization. In IRIS, the lattice pitch and fuel assembly overall dimensions are kept constant for interchangeability. At the same time the fuel rod diameter (and consequently pitch-to-diameter ratio, p/d, V_m/V_f, and neutron moderation) may be changed in the future to match a higher fissile content and/or MOX fuel. The initial lattice already has a somewhat increased V_m/V_f, to about 2. Compared to present PWRs, additional operational and safety margin is provided by a somewhat reduced average linear heat rate (~3 kW/ft).

Fuel lattice parameters for potential future reloads and the first core are compared in Table 6. They are selected for each fuel type to provide adequate neutron moderation, while maintaining other parameters (e.g., pellet diameter) in the desirable range. Thus, in the variable moderation approach, the increase in fissile content is matched by an adequate increase in moderation ratio, by adjusting the fuel rod diameter, while keeping the fuel assembly envelope unchanged. These advanced reloads can be envisioned to become available in the 2020s, as a higher burnup database becomes available and a higher fissile content becomes licenseable.

Table 6 IRIS Is Designed to Accommodate Core Upgrades

	Initial Core	Future UO_2 Upgrade	Future MOX Upgrade
Fuel Type	UO_2 <5% fissile	UO_2 >5% fissile	MOX >5% fissile
Fissile Content	4.95%	~8%	~10%
Core Lifetime	4-5 years	~8 years	~8 years
P/d	1.4	1.5	1.7
V_m/V_f	2.0	2.5	3.7

The IRIS capability of core interchangeability is one key reason for the expanded use of burnable absorbers in IRIS. In fact, extended cycle with associated excess reactivity and the capability of accommodating interchangeable cores impose more severe requirements on reactivity control in IRIS than in present PWRs. Therefore, it is desirable to use integral burnable absorbers, rather than solid rods that would occupy control rod guide thimbles, and to limit the control rods functions to shutdown and power shaping. The increased cycle burnup is helpful in one respect, since it leads to a higher depletion of burnable absorbers resulting in a reduced reactivity penalty, as compared to present PWRs.

Erbium is quite effective for longer life UO_2 cores (see Figure 12); as it was shown in Figure 8, IFBAs are not effective at higher burnups. On the contrary, due to its harder spectrum, MOX fuel would make IFBA more effective for long-term reactivity control, since the boron depletion rate is reduced. As shown in Figure 13, excess reactivity may be reduced from ~ 27% to only ~6-7% in the 3X-IFBA case.

The feasibility of advanced reload straight burn and higher burnup cores has thus been demonstrated, along with the capability of the IRIS design to accommodate them. No further work is therefore envisioned until the need arises.

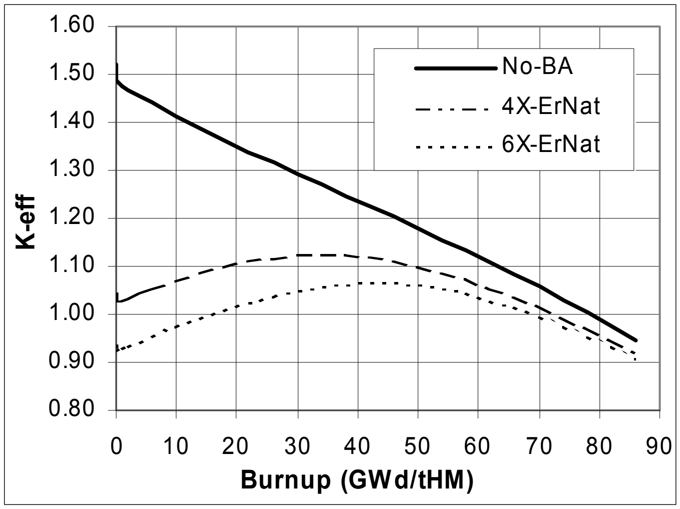


Figure 12 Effective multiplication factor as a function of burnup for IRIS UO_2 fuel, 9% enriched. Erbium burnable absorber.

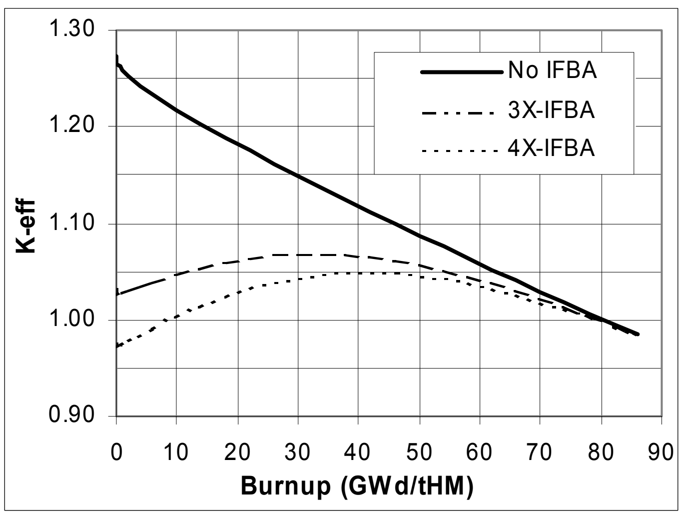


Figure 13 Effective multiplication factor as a function of burnup for IRIS MOX fuel, 10% fissile Pu. IFBA burnable absorber.

5.3 VERY LONG LIFE ADVANCED CORES (DEPLOYABLE AFTER 2030)

As seen before (Figure 4b) very high burnups, well in excess of 100,000 MWd/tHM, can be obtained with MOX fuel in tight lattices and higher fissile content. Straight burn core lifetimes of the order of 15 years are also possible. This is hardly surprising, based on the experience gained with fast reactors. A very tight lattice (approaching p/d ~ 1.0) reduces dramatically the coolant moderation and an epithermal spectrum results. The increased internal conversion ratio characteristic of the harder spectrum allows a rather flat reactivity profile with time, thus ensuring the long life burn. Again, hardly surprising since ultra long life cores of 15 to 30 years had been designed in the breeder program.

Development of these cores required addressing two difficulties, which are much more severe in a water-cooled integral reactor like IRIS than in a liquid metal cooled reactor:

- A positive void coefficient towards end of life. In sodium cooled reactors a positive void coefficient could be tolerated given the strong negative Doppler and overall power reactivity coefficient.

- The reduced heat removal capability in a very tight bundle is much more severe for a water cooled reactor than for a high conductivity liquid metal cooled reactor.

Fundamental development work to resolve the above difficulties was undertaken by three universities: the University of California at Berkeley (UCB), the Massachusetts Institute of Technology (MIT) and the Tokyo Institute of Technology (TIT).

UCB concentrated on the neutronics aspects; in the first year it investigated long life cores with small burnup reactivity swing, looking at UO_2 and MOX fuel, adoption of boiling spectral shifting with heavy water and zoned enrichment (see Reference 1). It was found that while a near zero burnup reactivity swing for 15 years of operation was readily achievable, attaining a negative void coefficient was quite more difficult. In the second year,[2] use of ^{232}Th instead of ^{238}U as fertile material was investigated, also with negative results. Further work was originally going to investigate the effect of layering the fuel both axially and radially, but in the third year the UCB effort was redirected to support the first core effort.[1]

MIT and TIT work in the first year[1] focused on assessing alternate exotic fuel geometries in lieu of the traditional cylindrical fuel rod. The starting point is rather simple: in a square rod array, and to lesser extent in a triangular array, a significant fraction of the coolant flows in the center of the subchannel, and thus it does not contribute much to fuel cooling. The ideal configuration is one where the coolant is equally distributed along the surface of the rod. Figure 14 shows the velocity distribution in a triangular cylindrical rods array and in a hexagonal/multi-lobe shaped fuel bundle. The much better coolant utilization in the latter, yielding a flatter temperature profile and lower peak around the rod circumference, is shown in Figure 15.

In subsequent years MIT continued to look at exotic fuel geometries, assessing pressure drop and critical heat flux correlation applicable to twisted hexagonal geometries, but mostly concentrated on the design of a tight lattice epithermal reactor with cylindrical rods.[2] Scoping neutronics, thermal hydraulic and economic assessment was performed, and the conclusion was that a tight core can indeed be quite attractive. The tight core allows a much higher power density and thus significantly reduces the capital cost per installed MWe. Even though the fuel cycle cost tends to be higher, there is potential to improve the overall cost of electricity.

To correctly model flow in tight lattice bundles, an approach capable of describing anisotropic turbulence is necessary. TIT has recently undertaken a fundamental study of turbulence models and has evaluated their capability of predicting experimental data.

[1] This included developing full 3-D Monte Carlo models of the core and neutron radial reflector to benchmark modeling of the latter in nodal theory models. Additionally, Monte Carlo simulations of the benchmark core configuration were performed. Finally, internal shielding was evaluated using Monte Carlo to validate discrete ordinates methods developed by ORNL.

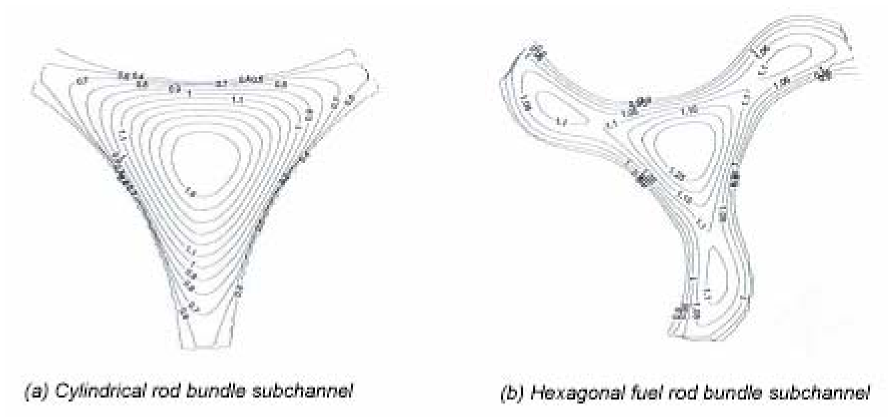


Figure 14 Velocity Profiles for (a) Cylindrical and (b) Hexagonal Multi-lobed Fuel Pin Bundle Subchannel

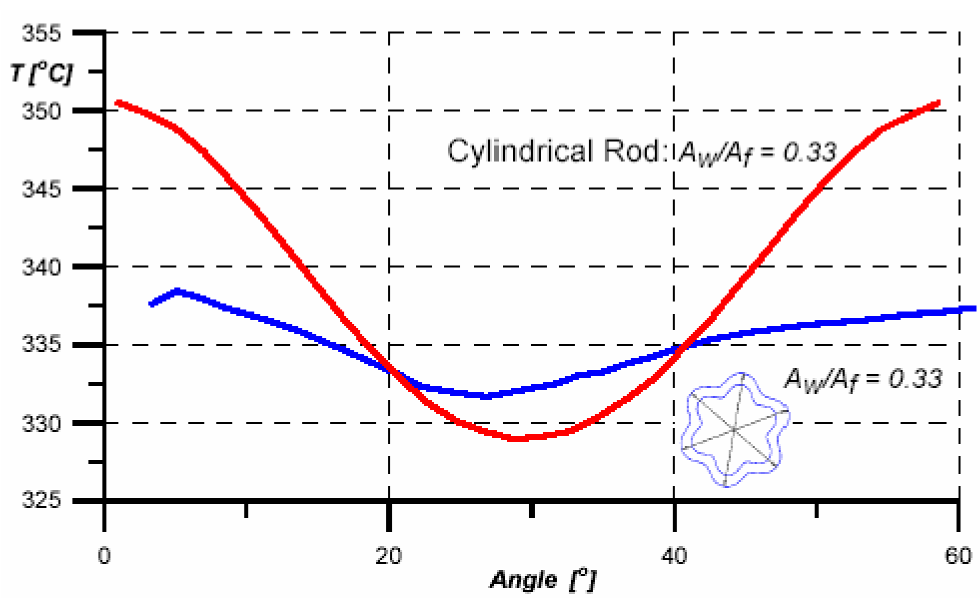


Figure 15 Cladding Wall Temperature Distribution at the Top of the Active Core for Cylindrical and Hexagonal Multi-lobed Rods

Non-linear models show the ability of reproducing to some extent the turbulence driven secondary motion, but substantial further work is needed.

Work on long life cores is currently not in the mainstream effort and it is performed at universities, as it appropriately fits a long term development. Obstacles are by no means solved; for example, besides the above need for flow modeling in tight bundles, additional work is still required to eliminate the positive void coefficient. However, it appears that substantial benefits can be obtained and appropriate technical solutions can be found.

6. PRIMARY SYSTEM INTEGRAL CONFIGURATION

Reported here is the current design of the IRIS reactor. Some iterations have occurred during the course of the project; for example, externally mounted, canned motor pumps like in AP600, were considered prior to adoption of the internal spool pumps and several types of steam generators were evaluated before deciding for the helical coil design. Also, the size and numbers of components have varied. For brevity and to avoid confusion, these iterations are omitted here.

The IRIS primary system includes: vessel and internals; steam generators; coolant pumps; pressurizer; neutron reflector; and, internal control rod drive mechanisms. They are discussed in the following sections.

6.1 REACTOR VESSEL AND INTERNALS

6.1.1 Vessel

The IRIS reactor vessel (see Figure 16) consists of a cylindrical shell made of several courses, a semi-spherically dished bottom head and a flanged and gasketed removable upper head. Stainless steel cladding of 6 mm minimum thickness covers the internal surface of the vessel. The reactor vessel size and configuration is dictated largely by the space required by the steam generators and internally mounted reactor coolant pumps. The reactor vessel design data are summarized in Table 7.

Table 7 IRIS Reactor Vessel Parameters

Overall length of assembled vessel-closure head	22.21 m
Inside diameter of shell	6.2 m
Nominal base metal thickness	280 mm
Minimum cladding thickness	6 mm
Design pressure	17.24 MPa (2500 psia)
Design temperature	343.3ºC (650ºF)
Vessel material	Carbon steel, SA 508, Gr.3, Cl.2
Cladding material	Stainless steel

The removable upper head of the vessel contains a bolting flange with 72 eight-inch studs and nuts. Two hollow, metallic O-rings form a pressure-tight seat in concentric grooves in the head flange.

The reactor coolant system is fully contained within the reactor vessel and is pumped in a closed circuit within the vessel (see Figure 16) with the exception of some auxiliary systems (e.g., the makeup and purification systems). The coolant passes upward through the core, turns radially outward at the top of the upper internals, flows up to the eight primary pumps, is pumped downward through the pumps and through the steam generators, down the annulus between the core barrel and the reactor vessel wall, then upward through the core support assembly. The reactor vessel cylindrical wall has eight steam generator feed water inlet nozzles located above the core level and eight steam outlet nozzles located below the vessel flange. Steam generator feedwater passes through the feedwater nozzles into the feedwater header, enters the steam generator

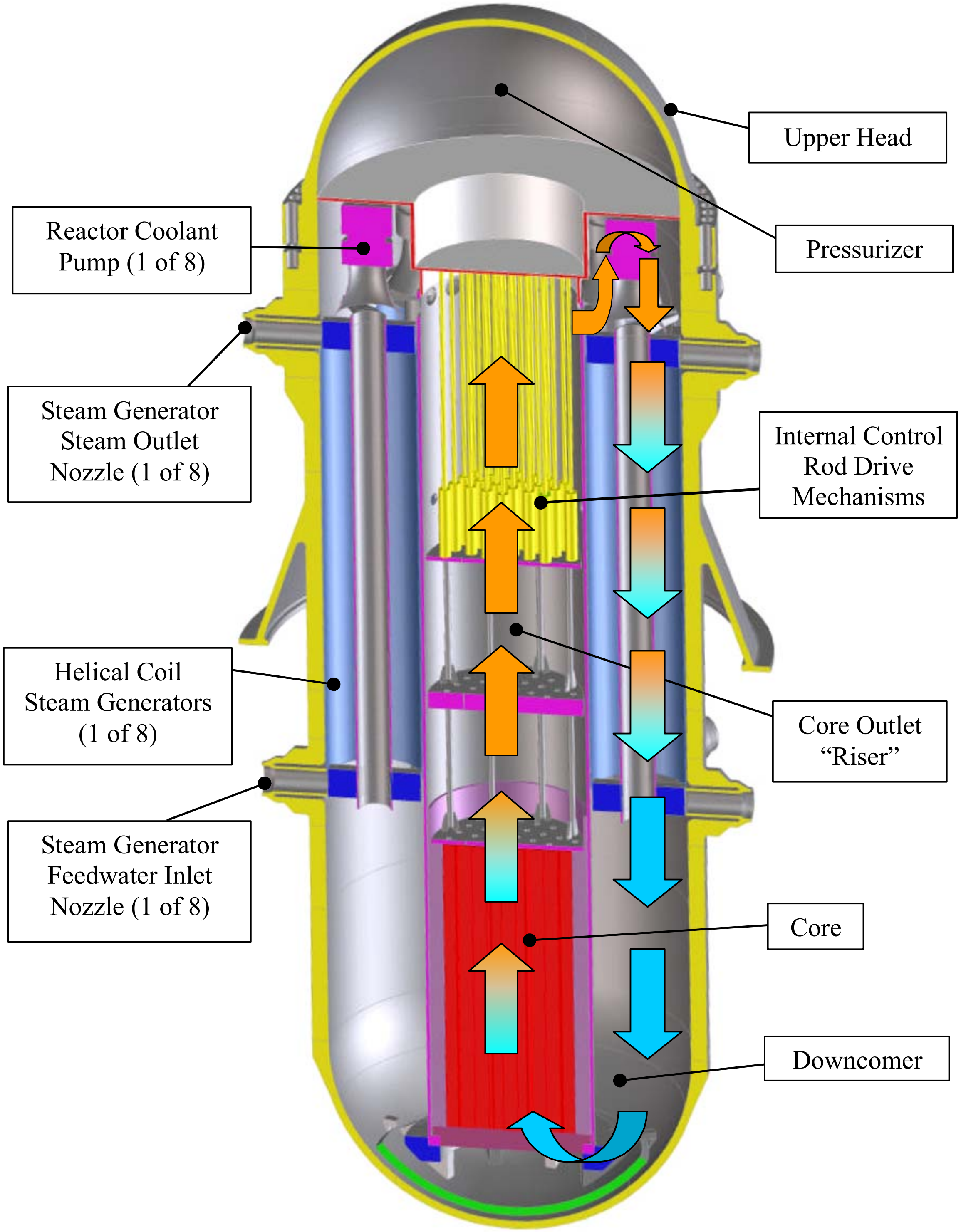


Figure 16 IRIS Integral Layout

tubes and flows upward inside the tubes, first being heated to saturation, then boiled, and subsequently heated to dry superheated steam, which then flows into the upper steam discharge header and out through the steam outlet nozzles to the turbines. Both the SG feedwater and steam headers attach directly to the reactor vessel inside wall and form the primary to secondary pressure boundary.

The reactor vessel and closure head are designed as a Class 1 vessel in accordance with the ASME Code, Section III. The current design life for the reactor vessel is 60 years, but its actual life is expected to be significantly longer because the radiation damage on the vessel is practically non-existent, as discussed in Sections 2 and 8. In general, all attachments and pressure containing parts have full penetration welds.

The reactor vessel support was initially by means of a cylindrical skirt welded to a “Y” forging between the lower cylindrical shell and the semispherical lower shell. The eventually adopted solution was to use a conical skirt welded to the cylindrical shell between the steam generator inlet and outlet nozzles, since it increases the natural frequencies of the vessel and reduces the vibration and dynamic interaction of the different components. This support is designed to restrain lateral, vertical, and rotational movement of the reactor vessel and still allow for thermal growth.

The dynamic evaluation due to seismic effects is one of the most important considerations in the design of the reactor vessel and internals supports. Another consideration in the design of the reactor vessel skirt is the therrmal stress due to the temperature gradient of the skirt at the attachment to the reactor vessel. Detailed thermal stress analysis of this area using finite-element techniques to determine primary plus secondary stresses of heatup and cooldown thermal transients will be performed. In addition, to provide good heat flow from the reactor vessel to the skirt a forged skirt attachment with full penetration welds and selective use of insulation in the crotch area will be used.

Large integral type forgings for the construction of big primary components of nuclear power plants have been used to reduce the manufacturing period, the length of welds and in-service inspection requirements. Figure 17 shows a possible course layout for the reactor vessel design. With the application of the shown integral type forgings, longitudinal welds are eliminated and circumferential weld seams are extensively reduced; thus, the following considerations apply:

– The use of integral type steel forgings for the fabrication of the IRIS reactor vessel enhances the structural integrity and facilitates fabrication and inspection, including in service inspection (ISI).

– In order to decrease the overall weight, the high strength SA 508, Gr. 3, Cl.2 is recommended for the reactor pressure vessel shell, flanges, and upper and lower heads.

All surfaces of the reactor vessel in contact with the reactor coolant are either clad with, or made from 300 series stainless steel and Inconel 690. Based on tensile and impact properties, Type SA 540, Class 3 is selected for closure studs, nuts and washers.

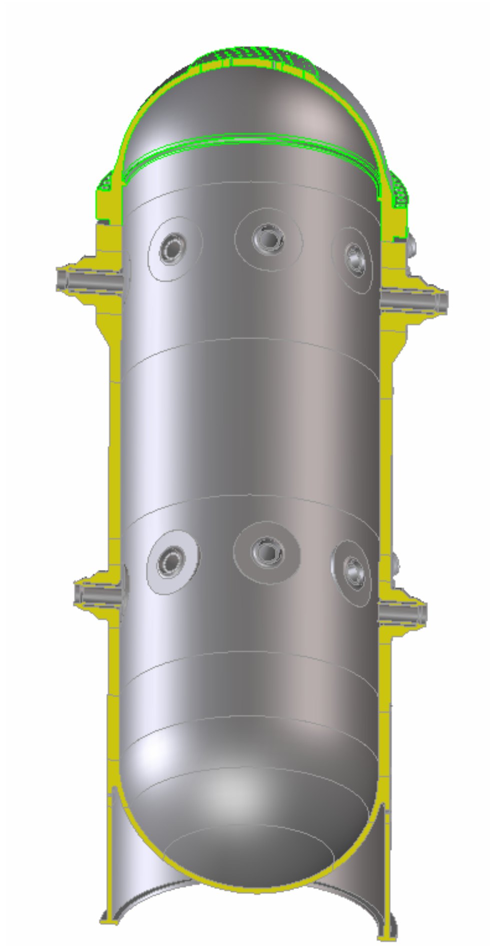


Figure 17 Layout of IRIS Vessel Courses

6.1.2 Internals

The IRIS reactor vessel internals (RVI) are similar to current PWRs in that they support the core, core barrel, control rods, control rod guide tubes and they also form the circulation path for the flow of coolant through the core. In IRIS, however, the RVIs provide the additional functions of supporting the internally mounted steam generators, reactor coolant pumps, control rod drive mechanisms, and radial shield plates, if needed. In addition, the IRIS RVIs must provide support for the pressurizer heater rods, and provide an extended length upper core barrel to form the core flow path. The internals are designed to withstand the forces due to weight, preload of fuel assemblies, dynamic loadings, vibrations and earthquake acceleration.

The IRIS reactor vessel and internals are designed to provide access to the fuel assemblies after removal of the closure head and upper internals. Also, the support structures of the recirculation pumps and the steam generators are being designed to permit removal of these components for out-of-vessel inspection and replacement.

The reactor internals are shown in Figures 18 and 19. They are divided into two parts:

1) The lower core support structure (including the entire core barrel and thermal shields)

2) The upper core support assembly

The major restraining and support member of the reactor internals is the lower core support structure, shown in Figure 18. This support structure assembly consists mainly of the core barrel, the radial reflector, the lower core plate, the triangular shaped core support members which are welded to the bottom head, and the core support ring which also functions as neutron shielding for a portion of the lower head. All the major components of this structure are supported at the bottom head; the lower end of the core barrel is restrained from transverse movement by a bolted connection to the support ring which rests on the (triangular) support members. Within the core barrel is the radial reflector, which fits inside the core barrel wall and form the enclosure periphery of the assembled core.

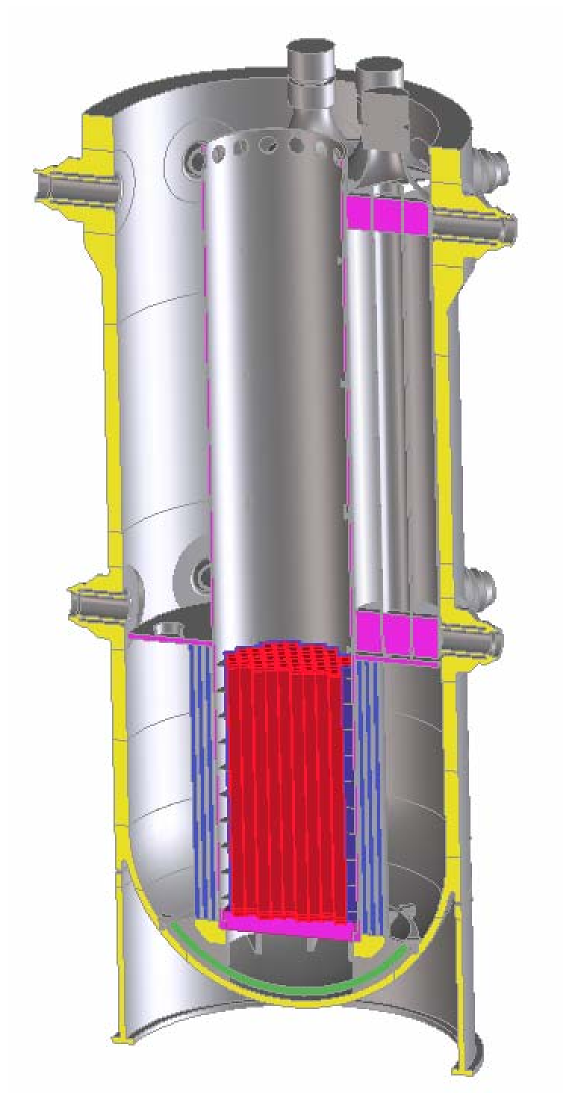


Figure 18 RPV and Lower Core Support Structure

The lower core plate is positioned at the bottom level of the core and provides support and orientation for the fuel assemblies. The lower core plate is perforated and contains the locating pins for the fuel assemblies. The lower core support structure (principally the core barrel) also serves to define the passage-ways for the primary coolant flow through the core.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, and earthquake acceleration are carried out by the lower core plate partially through the support ring to the lower (triangular) support members and to the bottom head. Transverse loads from earthquake acceleration, coolant crossflow and vibration are carried by the core barrel shell to be shared by the horizontal ledges, support ring and the vessel shell. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core support plate to the barrel wall and by a radial support-type connection of the upper core plate to slab-sided pins pressed into the core barrel.

With this design, the internals are provided with a support at the furthest extremity, with the core barrel bolted to the column supports, and may be viewed as a beam simply supported at the bottom. Radial and axial expansions of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design, keeping cyclic stresses in the internal structures within the ASME Section III limits, which essentially eliminates any possibility of failure of the core support.

The upper core support assembly (Figure 19) consists of the upper support plate, upper core plate, support columns, middle support plates and guide tube assemblies (not shown). The support columns establish the spacing between the upper support plate,

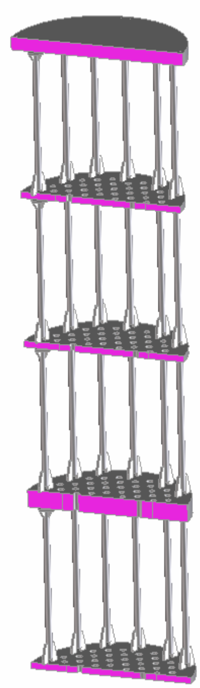


Figure 19 Upper Core Support Assembly

middle support plates and the upper core plate and are fastened at top and bottom to these plates; the support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting in-core and ex-core instrumentation conduits. The guide tube assemblies sheath and guide the control rod drive shafts and control rods, but provide no other mechanical functions; they are fastened to the lower middle support plate and are guided by pins in the upper core plate for proper orientation and support.

The main radial support system between the core barrel and the upper internals is accomplished by key and keyway joints. At equally spaced points around the circumference and coinciding with the level of each support plate, Inconel blocks are welded to the inside diameter of the core barrel. Each of these blocks has a keyway geometry; opposite each of these is a key which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction.

The upper core support assembly, which is removed as a unit during refueling operations, is positioned in its proper orientation with respect to the lower support structure by flatsided pins pressed into the core barrel which in turn engages in slots in the upper core plate. Slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies, and control rods is thereby assured by this system of locating pins and guidance arrangement.

6.2 STEAM GENERATORS

6.2.1 <u>Design</u>

Several configurations were examined for the IRIS steam generator (SG): straight-tube, U-tube, helical tube, C-tube, bayonet tube. Based on overall lifecycle costs, design and manufacturing experience, and high reliability, a helical-coil tube bundle steam generator was selected. The helical-coil tube bundle is a proven design that has operated in various reactors, including the French LMFBR Superphénix. There is also the ten years (1968-1979) operating experience of the PWR powered German nuclear ship Otto Hahn[10] with its 38 MW SG. The good experience of this nuclear ship did encourage the designer to carry out studies for larger-capability SGs of the same type up to a rated power of 190 MW.

The helical-coil tube bundle design is capable of accommodating thermal expansion without excessive mechanical stress, has high resistance to flow-induced vibrations, and is designed to have thermal performance second only to a straight-tube design (which was discarded because of the high loads due to thermal expansion caused by temperature transients, mainly compressive forces developed between the feed and steam headers).

In the early 90's Ansaldo designed the integral PWR 650 MWt ISIS (Inherently Safe Immersed System) reactor,[11] which is in many respects similar to IRIS. In particular, the ISIS SG is also an helical coil design and could be considered a reasonable reference design for IRIS. The innovative aspects of the ISIS SG were successfully tested in an extensive test campaign conducted on a 20 MWt full diameter, reduced height, test article shown in Figure 20. The test SG consisted of 50 tubes arranged in 5 rows of 10 tubes, each row forming – alternately – 5 clockwise and 5 counterclockwise coils. Performance characteristics (thermal, vibration, pressure losses) were investigated along with the determination of the operating characteristics domain for stable operation.

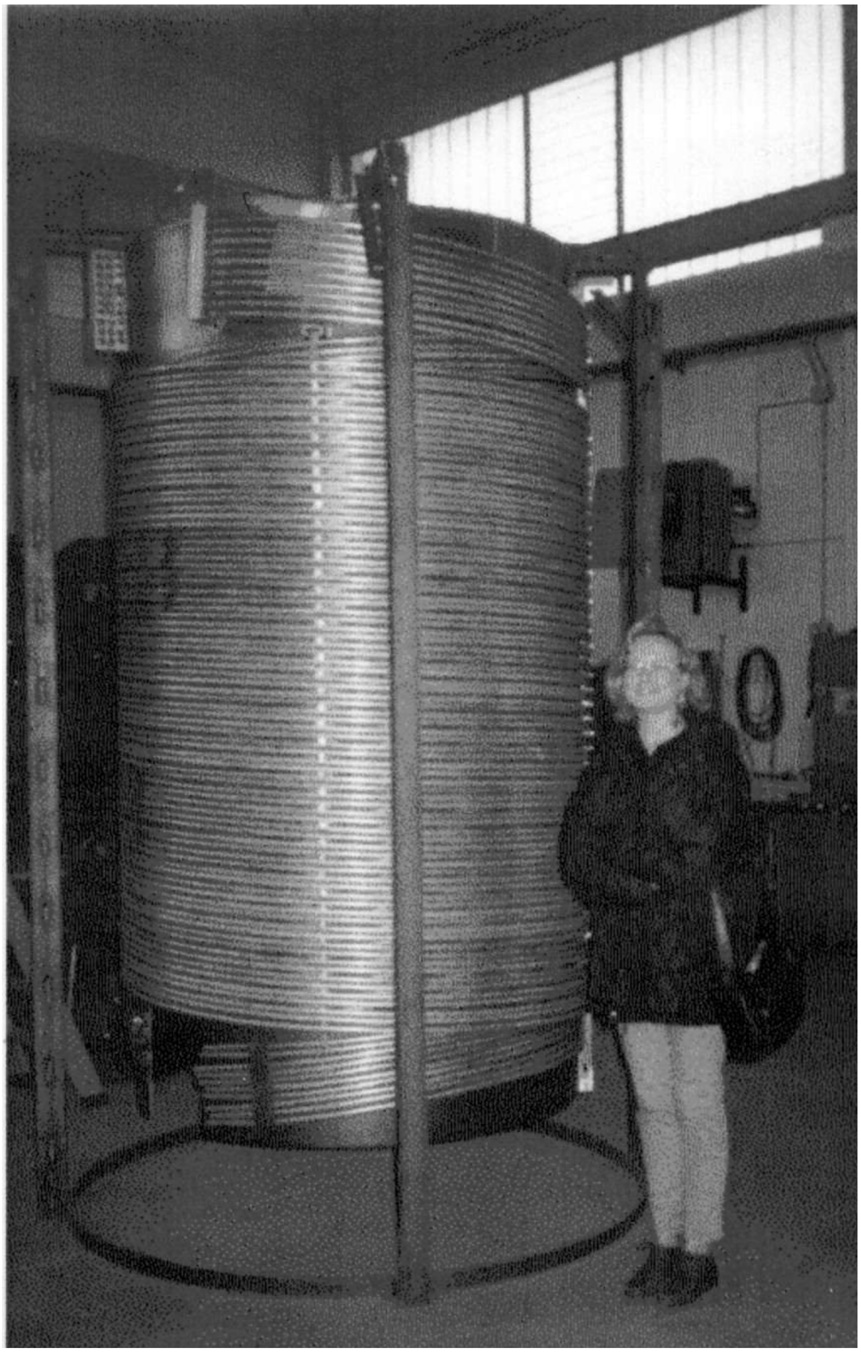


Figure 20 Mockup of IRIS Helical Coil Steam Generator

The IRIS design features eight identical helical coil steam generators modules (see Figure 21), completely separated and located in the annular space between the core barrel and the reactor vessel wall. This selection was based on the following considerations:

- Failure of one steam generator does not involve other units;
- At least four mechanically and functionally independent units are required.

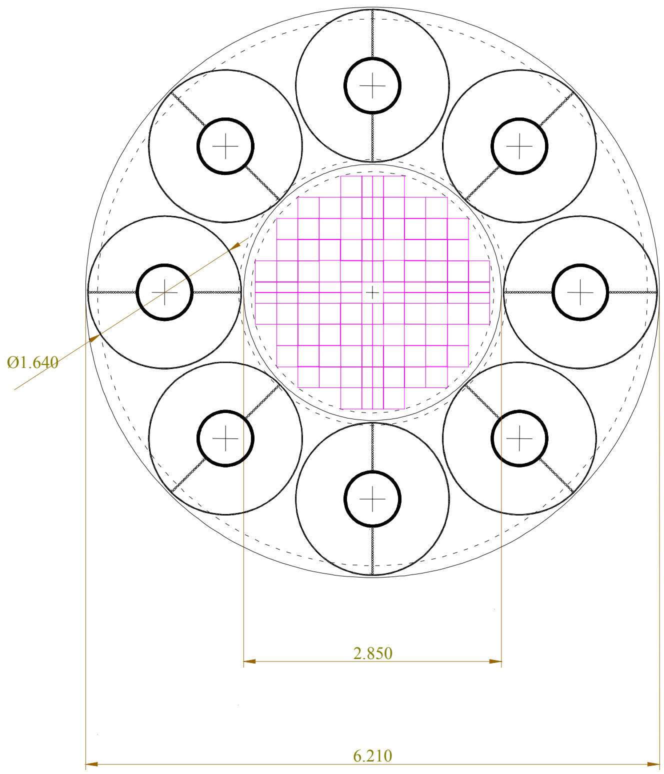
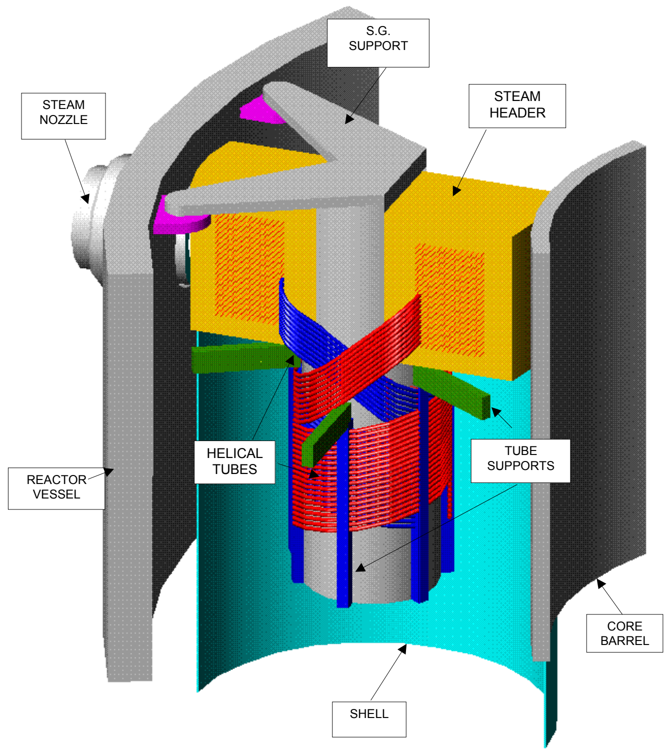


Figure 21 Layout of Steam Generator Modules

The adoption of the eight steam generators allows a modular construction of reduced-size components and limits the length difference between the various rows, so that it is possible to adopt single piece tubes of commercial length with no welding.

The SGs are once-through type and have the secondary side feedwater/steam inside the tubes and the primary side reactor coolant on the outside of the tubes. This means that the SG tubes are in compression and therefore are not subject to tensile stress corrosion cracking, which has been responsible for about 70% of SG tube failures in current PWRs. Even in the case of a tube failure, its consequences are much more benign than in loop PWRs, as it will be discussed in Section 10.3.2.

Each IRIS SG module (see Figure 22) consists of a central inner column which supports the tubes, with the lower feed water header and the upper steam header connected to



IRIS STEAM GENERATOR MODULE
Pictorial view

Figure 22 IRIS Steam Generator

the inside wall of the reactor vessel. The tube coils are .64 m in diameter and the helical tubes are arranged in annular rows. The tubes are connected to the vertical sides of the lower feedwater header and the upper steam header. The SG module headers are bolted to the vessel from the inside of the feed inlet and steam outlet pipe.

The steam, generated in the tubes, flows upward and exits through the upper header; feedwater enters the steam generator at the bottom header (at an elevation above the top of the core) through a feedwater nozzle. The tubes are fabricated of nickel-chromium-iron Alloy TT-690. Flow restriction orifices are provided at the tube inlet, to promote an even flow distribution through the tubes in the tube bundle and to avoid parallel channel instability. The required pressure drops for these orifices are of the same order as the tube pressure drops.

Key design parameters are reported in Table 8.

Table 8 IRIS Steam Generator Parameters (One of Eight)

Rated power	125 MW
Tube outside diameter	17.46 mm
Tube thickness	2.11 mm
Tube inside diameter	13.24 mm
Number of helical rows	21
Tubes number	656
Tube bundle average length	32 m
SG height (headers centerline)	7.9 m
SG overall height	8.5 m
Primary side inlet temperature	328.4°C
Primary side outlet temperature	292°C
Feedwater temperature	223.9°C
Steam temperature	317°C
Primary side pressure	15.5 MPa
Steam outlet pressure	5.8 MPa
Primary flow rate	589 kg/s
Secondary flow rate	62.5 kg/s
Primary side pressure loss	72 kPa
Secondary side pressure loss	296 kPa

Studies have been performed to confirm the applicability of the ISIS test data to the IRIS steam generator. The main results of the ISIS test campaign were:

- Absence of tube vibration
- Confirmation of the predicted thermal performance
- Identification of about 25% margin on the calculated primary side pressure losses
- Identification of the domain of stable operation as a function of: primary coolant inlet temperature; secondary coolant flow rate; secondary coolant inlet temperature; and, secondary coolant pressure.

6.2.2 <u>In-Service Inspection</u>

Since the SG is internal to the reactor vessel, a sound design, as discussed in the previous section, must be coupled to a sound in-service inspection system (ISI), diagnostics and prognostics. Discussed in the following is the proven ISI which was developed and implemented by Ansaldo jointly with Framatome for the Superphénix helical steam generator, using ultrasonic and visual inspection techniques. Following a convincing demonstration of performance (1993) and the final acceptance by EDF (1996), an inspection campaign was carried out (1997/98) for several SG tubes.

Figure 23 shows the mockup test apparatus of Ansaldo Energia for the Superphénix steam generator ISI. Figure 24 shows a proposed mechanism for probe introduction in the IRIS SG. It is possible to carry out ISI by simply removing the blind flange bolted on the steam nozzle, without removing the steam lines or having to operate from inside the reactor vessel.

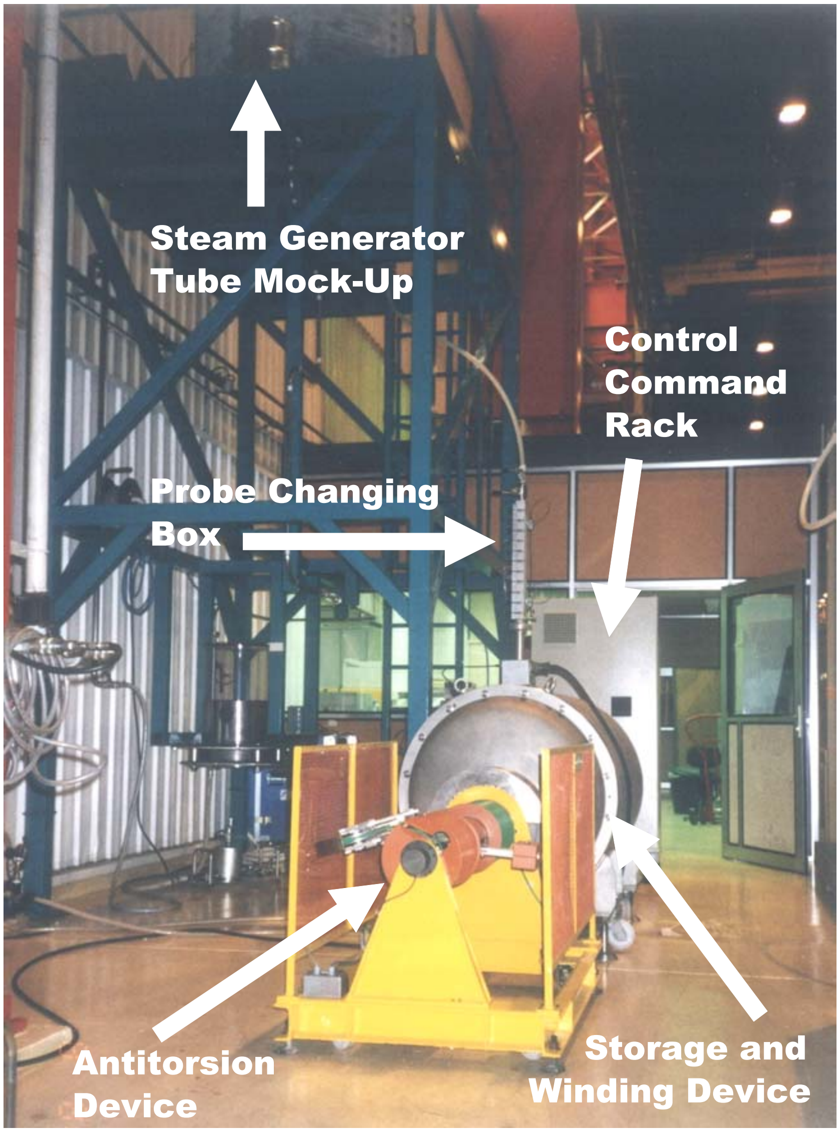


Figure 23 Mockup of Steam Generator Inspection System

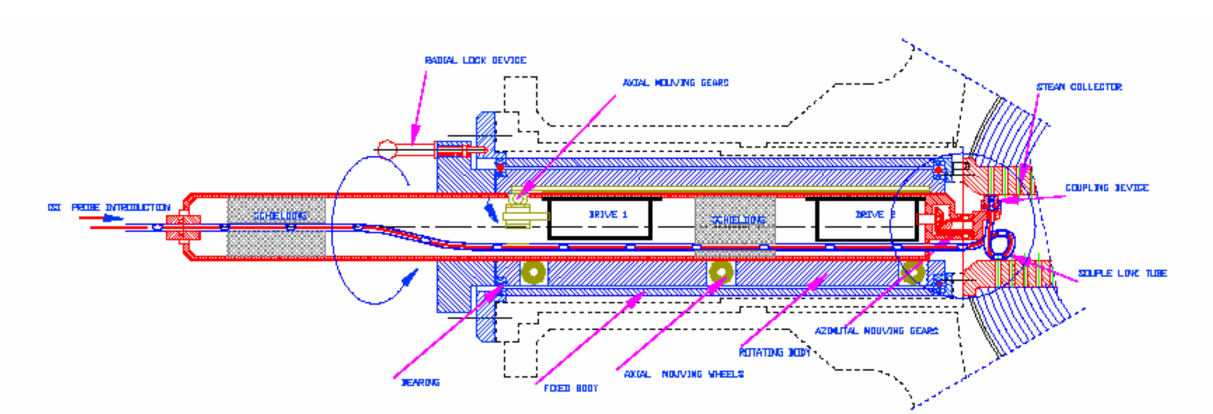


Figure 24 Scheme of the ISI Probe Assembly Bolted on the Steam Nozzle

In addition to the Ansaldo experience, Iowa State University/Ames Laboratories and the University of Michigan with Sandia National Laboratory are investigating through two NERI grants new monitoring technologies for material degradation and loss of integrity, as well as prognostic methods for predicting failures and set up preventive maintenance. A promising monitoring technology is EMAT (Electro Magnetic Acoustic Transducer) which can detect changes in tube diameter (thinning by corrosion or thickening by deposit), thus alerting plant operators to possible impending failures.

6.3 PRIMARY COOLANT PUMPS

The IRIS Reactor Coolant Pumps (RCPs) are of a "spool type," which have been used in marine and chemical plant applications requiring high flow rates and low developed head. Figure 25 shows the Integral Motor/Propeller (IM/P)^{TM}, which has been developed by Westinghouse ElectroMechanical Division (EMD), now Curtiss Wright, and is the fore-runner of today's spool type pumps. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor that carries high specific speed pump impellers. The spool type pump is located entirely within the reactor vessel, with only small penetrations for the electrical power cables and for water cooling supply and return. Further, significant qualification work has been completed on the use of high temperature motor windings. This and continued work on the bearing materials has the potential to eliminate even the need for cooling water and the associated piping penetrations through the RV. This pump compares very favorably to the typical canned motor RCPs, which have the pump/impeller extending through a large opening in the pressure boundary with the motor outside the RV. Consequently, the canned pump motor casing becomes part of the pressure boundary and is typically flanged and seal welded to the mating RV pressure boundary surface. All of this is eliminated in IRIS. In addition to the above advantages derived from its integral location,

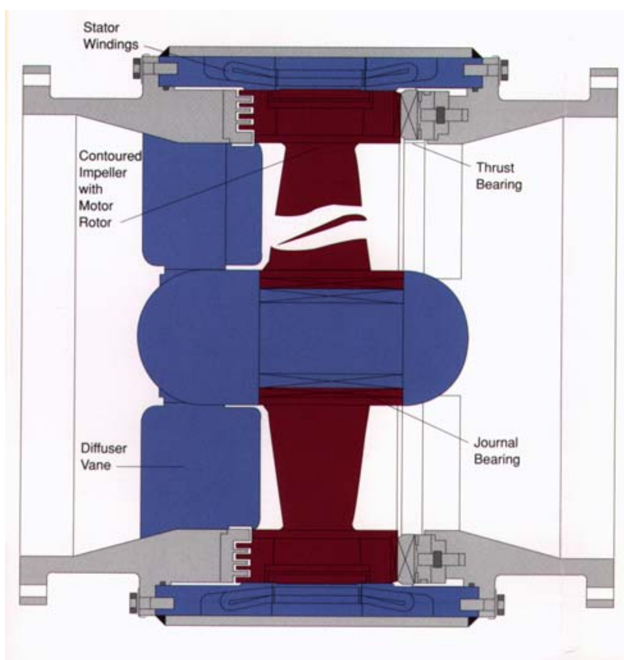


Figure 25 Westinghouse EMD IM/P Basic Components

the spool pump geometric configuration maximizes the rotating inertia and these pumps have a high run-out flow capability. Both these attributes mitigate the consequences of Loss-Of-Flow Accidents (LOFAs). Since their primary use is in hazardous chemical environment, the spool pumps have been designed not to require periodic maintenance. Because of their low developed head, spool pumps have never before been candidates for nuclear applications. However, the IRIS integral RV configuration and low primary coolant pressure drop can accommodate these pumps and take full advantage of their unique characteristics and therefore IRIS is the first commercial reactor design to utilize fully internal primary RCPs. Their use in non-nuclear applications is proven, however qualification for nuclear applications is necessary. Testing of the insulation and bearing systems have been conducted at 500°C (932°F); additional tests, including vibration, thermal cycling, and accelerated aging are planned.

6.4 PRESSURIZER

The reactor vessel upper head doubles up as the IRIS pressurizer. The IRIS solution is similar to that employed in loop PWRs: water-steam system, with the vapor formation (pressure control) accomplished by electric heaters. The self-pressurization at saturation conditions as adopted in the Otto Hahn or CAREM was not considered, in order to keep the reactor coolant at subcooled conditions to increase DNB margins and ensure that the reactor pumps, located near the pressurizer, maintain adequate suction head. Similarly discarded was the gas-supported pressurizer concept adopted in SMART[12] to avoid the problem of N_2 absorption in hot water, with associated accumulation in the colder regions.

The IRIS pressurizer region (see Figure 26) is defined by an insulated, inverted top-hat structure that separates the circulating reactor coolant from the saturated pressurized water. The functions of this structure include: (a) preventing the head closure flange and its seals from being exposed to the temperature difference between the reactor and pressurizer water, thus reducing thermal stresses and maintaining sealing tightness; (b) effecting a thermal insulation to minimize heat transfer and maintain an adequate saturated water layer within the pressurizer; (c) providing structural support for the core instrumentation and heaters; and (d) providing the communication flow paths between the reactor and pressurizer for the surge flows.

The closure head, as a part of the pressure retaining wall of the reactor pressure vessel, is designed as a Class 1 vessel according to the ASME Code Section III.

The key performance parameter of a PWR pressurizer is the ratio between pressurizer steam volume and reactor thermal power, which represents the pressurizer capability of reducing the rate of pressure increase during heatup transients. For IRIS this ratio is 3.4 times greater than a conventional two-loop PWR and more than five times greater than AP1000. In fact, the IRIS steam volume of ~ 50 m^3 is about 1.6 times higher than the AP1000 pressurizer steam space, while IRIS has less than one-third the core power. Thus, enough margin exists, so that IRIS does not need the pressurizer spray system used in loop PWRs to prevent the pressurizer safety valves from lifting during heatup transients.

Annular heaters are located in the top hat region to create and maintain the saturated water layer and to produce enough steam to prevent a pressure decrease during increases in plant power. To completely eliminate upper head penetrations (CRDMs are

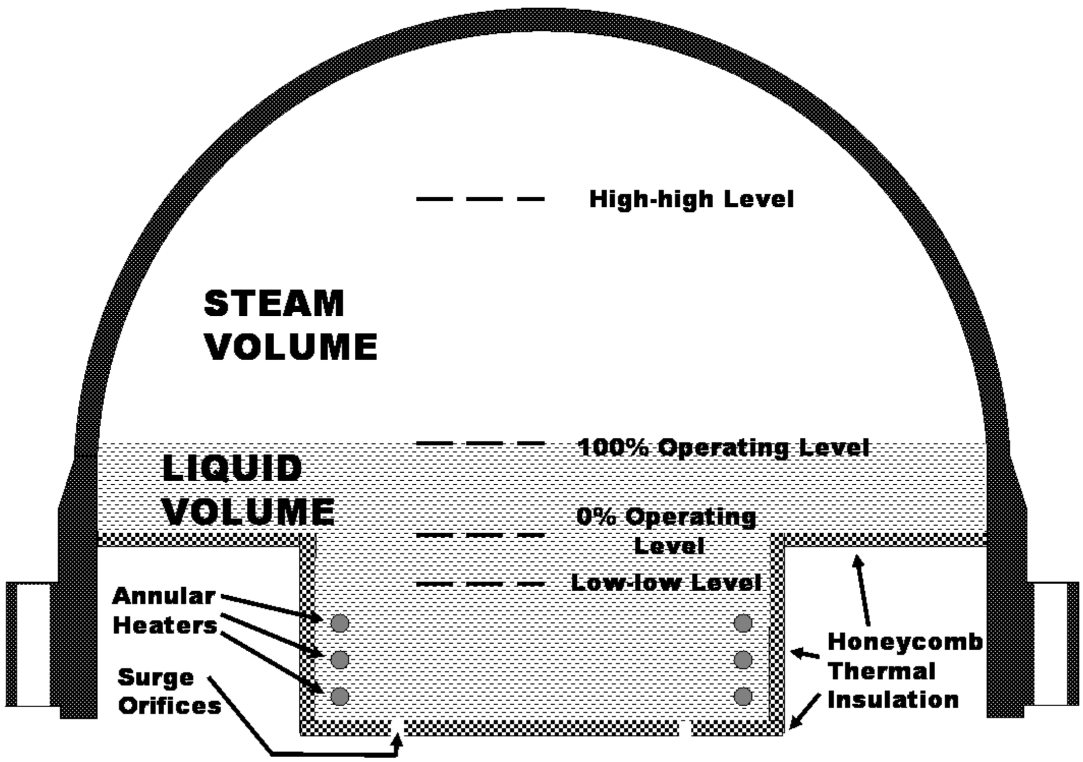


Figure 26 Pressurizer

also internal to the vessel, see Section 6.6), the heaters are fully contained in the pressurizer and are controlled via electrical cable.

6.5 NEUTRON RADIAL REFLECTOR

IRIS features a stainless steel radial neutron reflector (see Figure 27) to lower fuel cycle costs and to extend reactor life. The reflector reduces neutron leakage thereby improving core neutron utilization, and enabling extended fuel cycle and increased discharge burnup. It also has the added benefit of reducing the fast neutron fluence on the core barrel, and, together with the thick downcomer region, it significantly reduces the fast neutron fluence on the reactor vessel, as well as the dose outside the vessel to the extent of yielding, for any practical purposes, a "cold" vessel. This has obvious beneficial impacts on costs (very long life vessel, no need for the embrittlement surveillance program, reduced biological shield), operational doses, and decommissioning, as it will be discussed in more detail in Section 8.

6.6 CONTROL ROD DRIVE MECHANISMS

As mentioned in Section 2, IRIS will feature internal Control Rod Drive Mechanisms (CRDMs) for their significant operational and economical advantages. Two alternatives have been extensively studied around the world: electromagnetic and hydraulic

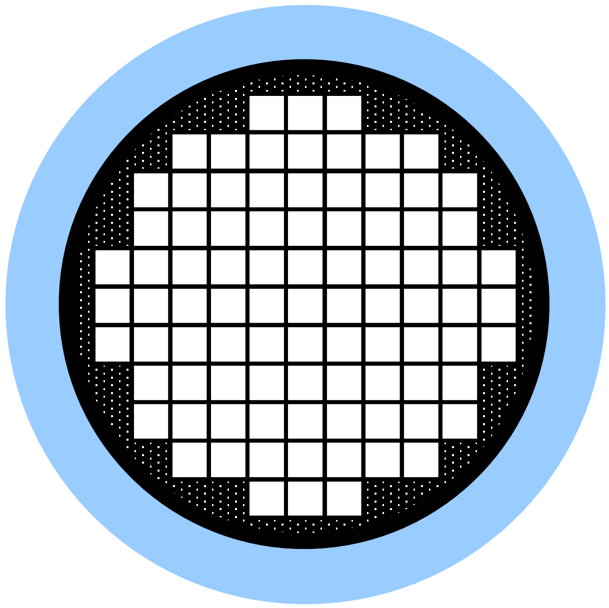


Figure 27 Neutron Radial Reflector

CRDMs; they are briefly discussed in the following, along with a recently proposed, different concept of control rods.

6.6.1 Electromagnetically Driven Internal CRDMs

This system has been and is being vigorously pursued in Japan. A CRDM that is located within the reactor vessel has been developed and patented by JAERI and MHI for the MRX[13,14] a small, pressurized-water, integral type reactor for nuclear propulsion of surface ships (see Figure 28). The rationale for the internal drive mechanism was to provide a compact design and to remove the possibility of rod ejection accidents, goals which are obviously of great importance in naval applications. The MRX however still retains some upper head penetrations.

This CRDM (see Figure 29) is driven by a canned, direct current, electric motor that rotates roller-nuts in contact with a lead screw portion of the control rod drive shaft. The roller-nuts (see Figure 30) which move the drive shaft are mounted in a split housing which can separate (open) when a latching magnet is de-energized. This results in the release of the drive shaft and dropping of the control rod. The latch magnet can be re-energized to close the separable roller-nut housing, so that the rollers are re-engaged on the lead screw (upper drive shaft), to resume their normal function. The CRDM materials were selected for projected compatibility with a high temperature, high pressure, radiation environment. Similarly to current power generation PWRs, the mechanisms are attached to the inner surface of the reactor vessel head and the CRDM drive shaft can be separated from the control assembly. This allows the CRDM to be removed along with the head, leaving the control assembly in the core during refueling operations.

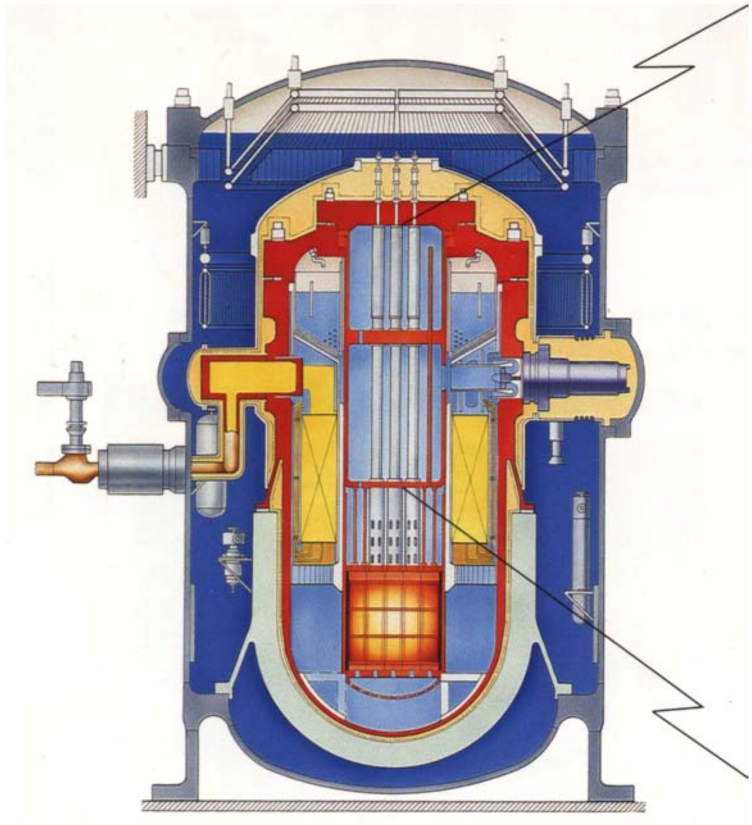


Figure 28 MRX (Advanced Marine Reactor) Incorporates an Internal Control Drive System Patented by MHI

Notable features of this system are as follows:

– Synchronized, canned motor with a permanent magnet encapsulated in the rotor. This design provides ease of selection and control of normal rotation, reverse rotation, speed, and stop (holding function).

– An integrated rod position indicator designed for high temperature endurance, that provides high accuracy rod position indication.

– A separating ball-nut housing in the drive mechanism that enables insertion and withdrawal operations and can drop the rod (scram function) at any rod position.

This CRDM has been designed for typical PWR operating conditions:

– Operating Temperature (°F/°C)	608/320
– Operating Pressure (psig/MPa)	1740/12.0
– Lifting Force (lb/kg)	440/200
– Stroke Length (in/mm)	55/1400
– Driving velocity (in/mm per minute)	12/300
– Scram Time (sec)	1.4
– O.D. (in/mm)	8/205

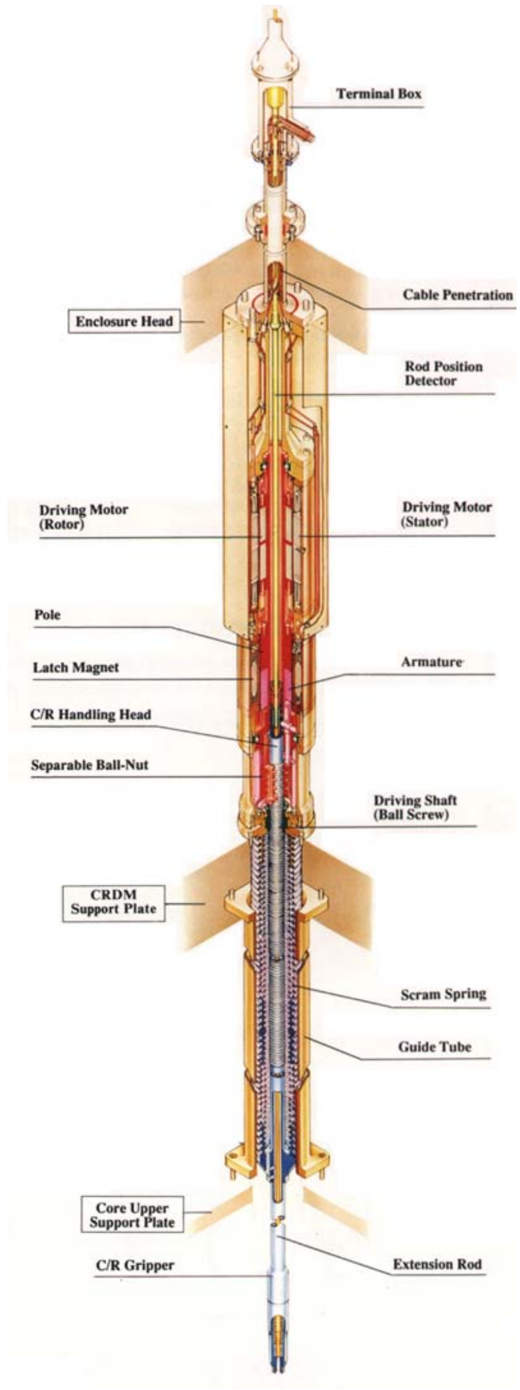


Figure 29 In-vessel CRDM Designed by MHI

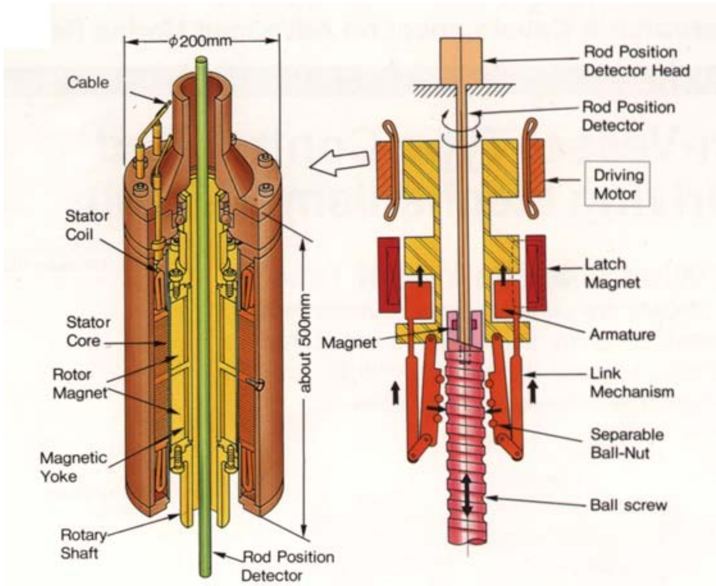


Figure 30 The MHI Electromagnetic Driven System

An extensive development program has been conducted since the early 1990's to address most of the uncertainties affecting the design. Components testing at room conditions and hot functional testing were conducted confirming the satisfactory behavior of motor, latch and trip characteristics. Magnet, bearing and structural materials testing was performed. Rod position indicators were tested.

MHI is currently adapting this CRDM design to their 300 MWe integral reactor IMR.

On the other end of the plant size spectrum, Toshiba, together with TEPCO and the University of Tokyo, has been pursuing an internal CRDM system for next generation 1700 MWe BWRs[15] (see Figure 31). This program is aimed at replacing the typical BWR bottom mounted CRDMs with internal ones mounted above the core, to reduce the height of the vessel and containment.

The key components of the BWR internal CRDMs (see Figure 32) are:

- Heat-resistant motor for positioning of the control rods
- Heat-resistant solenoid drive latch mechanism for gravity driven scrams
- Electromagnetic power coupling for signal and power transmission across the primary pressure boundary

A development program is under way addressing: the high temperature (600°C) behavior of the ceramic, insulated, heat and radiation resistant motor and coils, driving mechanism, and latch magnet; the durability of CRDM motor and roller nut ball bearing in high pressure and temperature reactor coolant; structural integrity and flow instability due to two-phase flow at the core exit. Material tests and proof-of-principle mechanical tests indicated the feasibility of the concept.

Every indication points to the fact that electromagnetically driven internal CRDMs are going to be adopted in one or more Japanese reactor types.

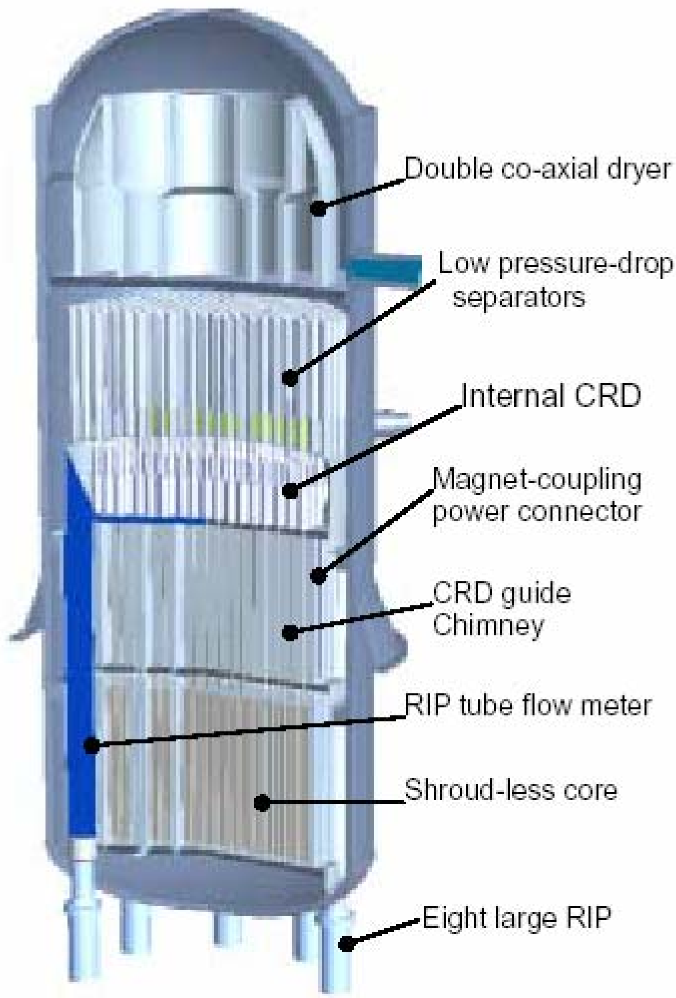


Figure 31 3D-CAD View of the Next Generation BWR

6.6.2 Hydraulically Driven Internal CRDMs

While the electromagnetically driven CRDM is a design pursued exclusively in Japan, the hydraulically driven CRDMs have been investigated in various countries. The first reported investigation of this concept is by Kraftwerk Union AG (KWU) and Siemens for the KWU 200 MWth reactor.[16] A schematic of the proposed design is shown in Figure 33. The hydraulic circuit is fed from the primary water through a pipe connected near the top of the reactor vessel. The pumps, which provide the hydraulic force for the rod movement, are located in the lower part of the vessel, in order to have a sufficient intake head to avoid any cavitation. In the Kraftwerk Union AG-Siemens reactor, the pump has a flat head-flow characteristic to guarantee stable operating conditions, with either one or two pumps operating in parallel (to provide redundancy).

Schematically, the hydraulically driven system consists of a piston, to which the control rod, i.e. the neutron absorbing structure, is fixed, moving inside or outside a fixed cylinder. The movable and fixed components are machined to obtain a periodic hydraulic profile between the piston and the cylinder, thus generating variable pressure

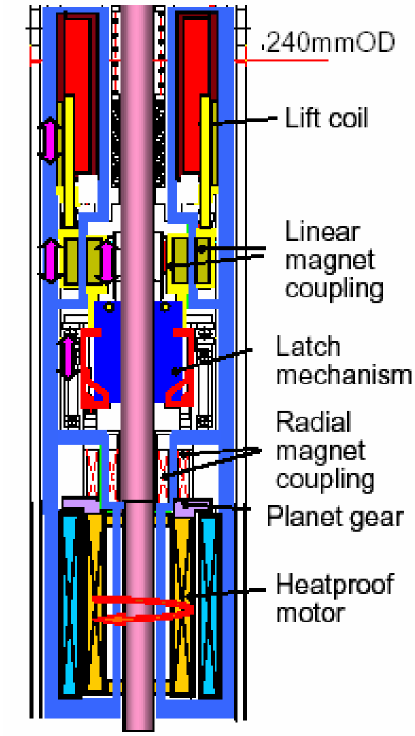


Figure 32 Schematic Drawing of the Toshiba/TEPCO Internal CRDM

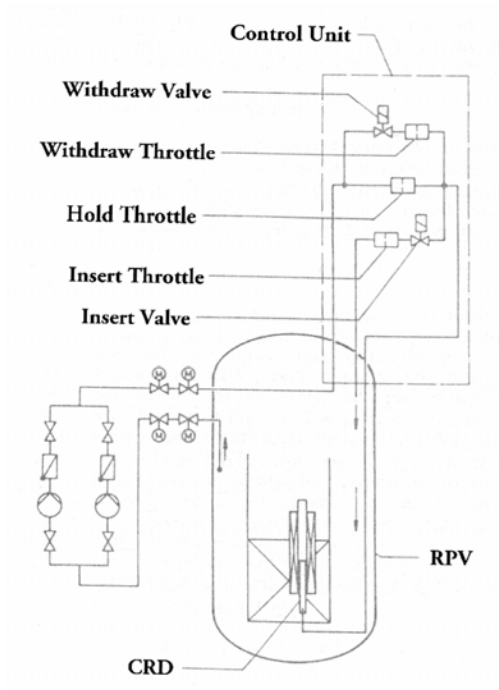


Figure 33 KWU200 Hydraulic Control Rod System

losses as the piston moves upward or downward and consequently causing notch movements of the control rod. Figure 34 shows the piston/cylinder geometry for the KWU200 and two subsequent designs, the Argentinean CAREM[8] and the Chinese NHR-5.[17]

The KWU and NHR designs have a movable hollow piston with a fixed cylinder, while CAREM is the reverse. KWU200 and CAREM are characterized by identical periodic profiles for both the piston and the cylinder surfaces, while NHR-5 accomplishes a periodic profile for the pressure losses via periodic set of holes in the fixed cylinder and one set of holes in the piston.

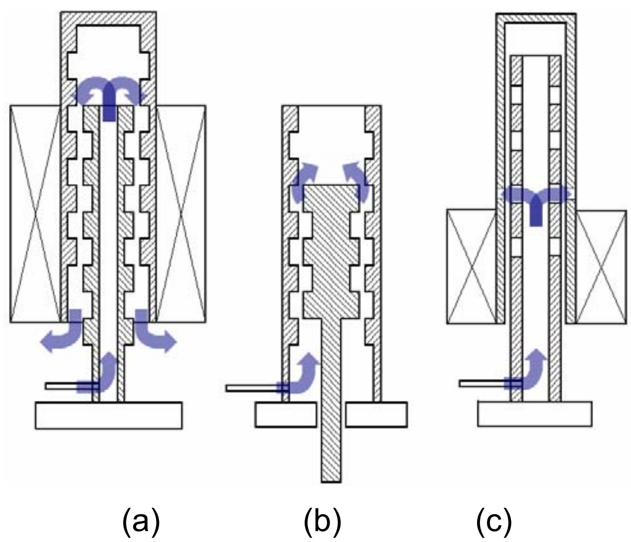


Figure 34 Hydraulic Drive System for (a) KWU200, (b) CAREM, (c) NHR-5

The KWU reactor was proposed for district heating applications and was dropped from consideration many years ago. A similar concept was however actually built and installed in China in the NHR-5 (Nuclear Heating Reactor – 5 MWt) operating since 1989 at Tsinghua University in Beijing.[17] This is a nuclear heating reactor operating at low pressure – low temperature conditions. It is reported[18] that the reactor has operated successfully. In fact, a hydraulically driven control rod system is currently being designed for the commercial sized 200 MW reactor NHR-200.[19] The CAREM system has been successfully tested and CAREM has been announced as ready for construction.

6.6.3 Liquid Control Rods

This system[20], proposed and patented by a small French company, MP-98, uses liquid neutron absorber (In-Cd eutectic alloy with a melting point about 120˚C), which replaces the control rods. The absorbing liquid is stored in tanks located above the core, and the tubes are filled with the absorber or flushed out by the applied helium pressure, as schematically illustrated in Figure 35.

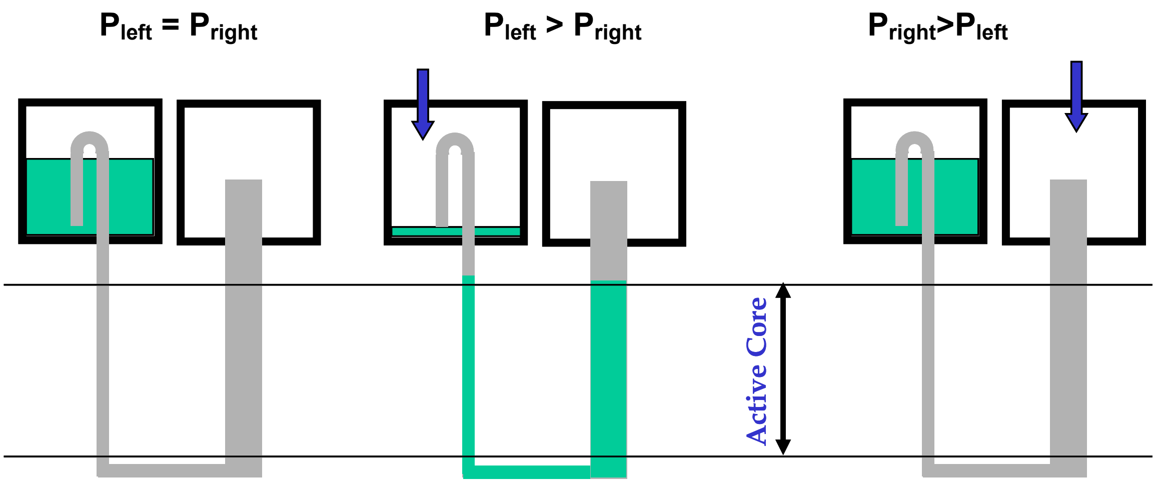


Figure 35   Principle of Operation of Liquid Control Rods MP-98

The tanks and tubes are integral part of each fuel assembly (Figure 36). Helium supply and control valves are located outside the pressure vessel, and pressurized helium is provided through the conduits embedded in the specially designed upper internals, via the top connection caps. A functional layout is shown in Figure 37.

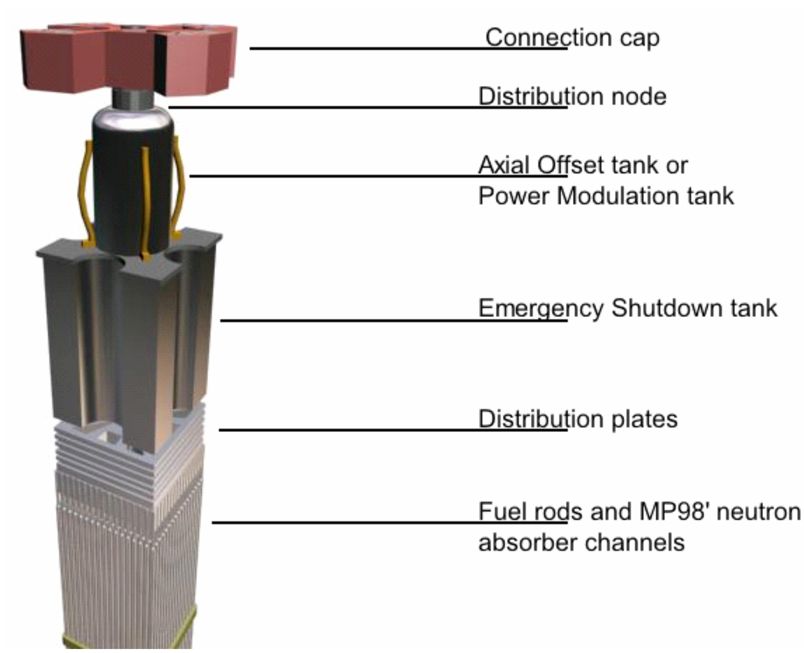


Figure 36 Fuel Element with Liquid Control Rods MP-98

The absorbing liquid is solid and “inserted” in fuel assemblies during transportation or refueling, thus inherently ensuring sub-criticality; it becomes liquid under operating conditions. The 24 tubes within each fuel assembly are divided into two groups (8 and

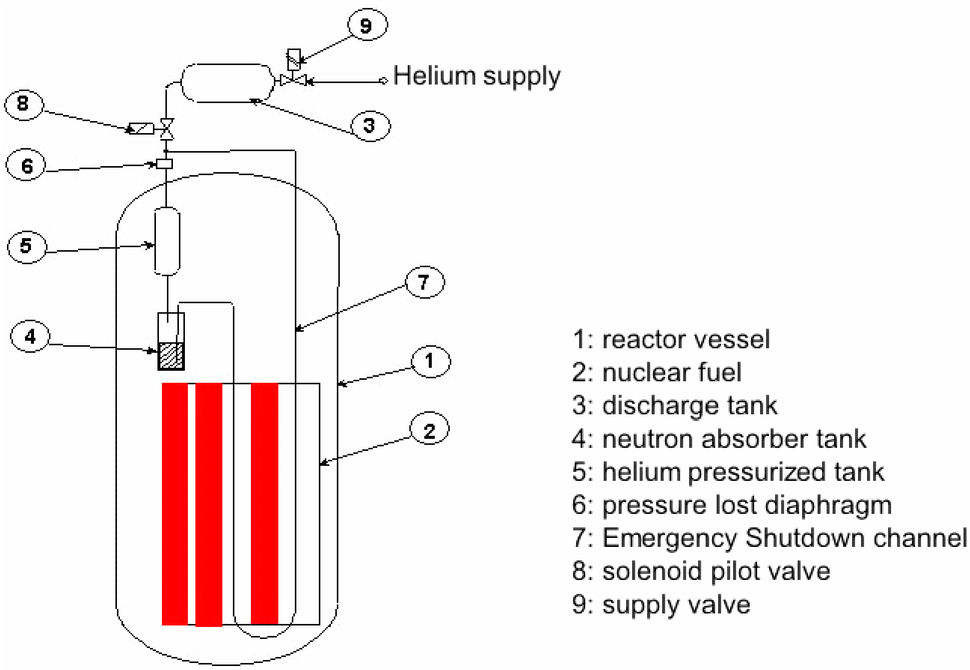


Figure 37 Functional Layout of the MP-98 System

16 liquid control rods), each group with a different function (e.g., shut-down and power-shaping rods). Since each fuel assembly may contain liquid control rods, this system offers a larger total rod reactivity worth and a more uniform absorber distribution than other control rod systems, and is suited for implementing soluble-boron free core operation. It may also provide better control for long cycle, higher enrichment, and/or MOX cores.

6.6.4 Assessment

The liquid rods offer many attractive features, most prominently the elimination of soluble boron, but are the farthest away in terms of actual development and testing. Since they are promoted by a very small company, there is also a high degree of uncertainty whether such development will actually take place. The IRIS project is in contact with MP-98 and will continue to monitor closely the progress of the concept.

The hydraulic drive is the only one which has been implemented in an operating reactor. It has the significant advantages that the operational controls (pumps and valves) are outside the vessel and that it does not require materials development. The IRIS project is very familiar with this concept and has performed development work on its own (see Section 7). A concern is the possibility that the delicate piston/cylinder geometry profile, and thus the rod movement, can be affected over lifetime by erosion/corrosion/ deposition; not much geometric variation is required to affect the hydraulic profile. The major concern however is that all the plants considered for the hydraulic drives are of small or very small size, at most a few hundred MWt rather than the 1000 MWt of IRIS. The engineering of the "plumbing" for a relatively large core and their reliability/accessibility/maintenance are question marks.

The electromagnetic drive presents the least deviation from the current technology, “merely” relocating essentially the same mechanisms (roller nut drive motor and latch coil) from outside to inside the vessel. A very comprehensive development program has been performed and there is a substantial industry interest for their deployment. The major question mark is their long term reliability; even though the radiation field is low and tests have been performed, still it is a more severe environment (significantly higher temperature and pressure) than previous applications. Also, remote control and positioning need to be qualified; however, the technology is mature.

7. INVESTIGATION OF HYDRAULICALLY DRIVEN CRDMs

As seen in Section 6.6.2, the hydraulic drive is one of the technical solutions which have been considered for the IRIS internal CRDMs; to have a better understanding of its feasibility, capabilities, and challenges, POLIMI has performed both analytical and experimental investigations of the concept. Starting from the current state-of-the-art of the piston-cylinder configuration shown in Figure 34, POLIMI selected a geometry similar to that of KWU 200, but with a profile less prone to erosion/deposition and a 50 mm total pitch (see Figure 38).

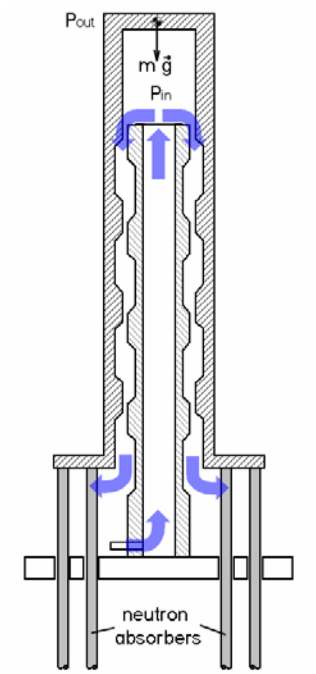


Figure 38 Proposed Configuration for IRIS Internal CRDM Hydraulic Drive

The main components of the control system to achieve a steady state configuration are schematically reported in Figure 39, where an electric equivalent circuit describes the concept. A centrifugal pump is needed to supply the equilibrium pressure ($P_{equilibrium}$) under the piston, as required to balance the weight of the control rod. In a steady state position, the pressure losses of the internal periodic profile of the coupled piston-cylinder surfaces are equal to the whole equilibrium pressure (ΔP_{rod}).

In order to simplify the control strategy, the pump operates at fixed revolution speed. For piston movement operations, the pump head supplies an extra pressure, which in steady state is dissipated through an orifice inserted in series into the hydraulic circuit (for simplicity represented by a control valve - Hold Valve).

The pressure losses of the control rod are a periodic function of the rod position and a typical relationship is reported in Figure 40, where an equivalent pressure loss coefficient β_{rod} is shown as a function of the rod position: the maximum value for the

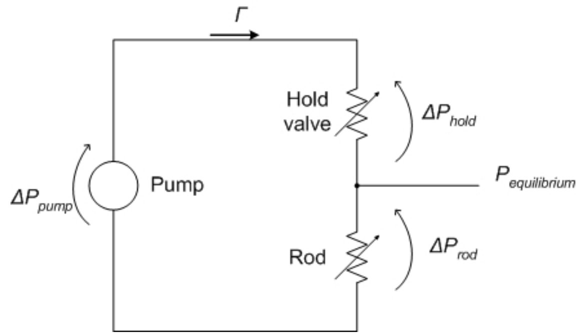


Figure 39 Equivalent Electrical Scheme of the HDCR System, for Steady State, Equilibrium Positions

pressure losses is achieved when the piston and the cylinder internal profiles are fully coupled, i.e. there is a minimum gap between the piston and the cylinder surfaces.

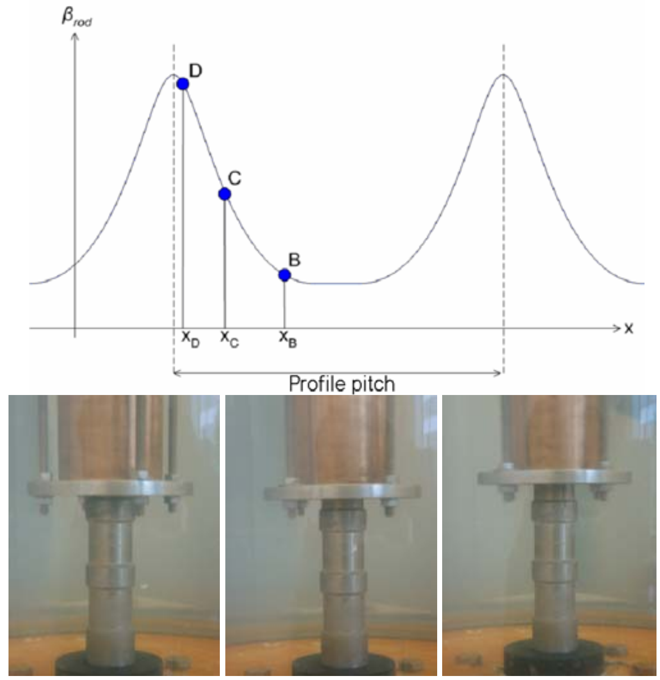
Different steady state, equilibrium positions can be reached and maintained by simply adjusting the opening of the Hold Valve, i.e. the hydraulic resistance. From Figure 40, with configuration (C) as reference, an increase in the Hold Valve pressure losses leads to a general decrease in the circuit flow rate, turning into a corresponding decrease in $P_{equilibrium}$ (control rod sustaining pressure). Hence the control rod has to travel to a new equilibrium position, where the pressure loss coefficient increases in order to balance the Hold Valve pressure loss: this can be accomplished only by moving downwards, towards position (D).

The same reasoning applies to a Hold Valve opening with respect to configuration (C): the circuit flow rate increases since the total resistance decreases, the sustaining pressure increases, the control rod moves upwards up to a new equilibrium position (B) where the rod ΔP_{rod} balance the Hold Valve ΔP_{hold}.

Computational Fluid Dynamic (CFD) analyses with the FLUENT code have been performed for a steady state simulation of the pressure losses distribution within the coupled piston-cylinder profile, at different relative positions. A total of 26 control rod different positions, within one pitch range (50 mm), were simulated: from complete coupling of the edge profiles, leading to the maximum Δp value, up to complete uncoupling of the edges.

The results are summarized in Figure 41 where the equivalent β_{rod} values are reported as a function of the relative positions of the piston and cylinder profiles, showing the same relationship as assumed in Figure 40. Equilibrium positions are on the downside part of the β_{rod} profile, while both the plateau and the upside parts represent unstable points.

The proposed system is similar to those adopted in the KWU 200 and NHR design. In addition to the steady state components shown in Figure 39, i.e. a pump, an orifice or a control valve rod, the system requires other components to allow the movement of the control rod, i.e. the withdrawal and the insertion steps. The whole command and control circuit is shown in Figure 42.



(D) (C) (B)

Figure 40 Different Steady State, Equilibrium Positions

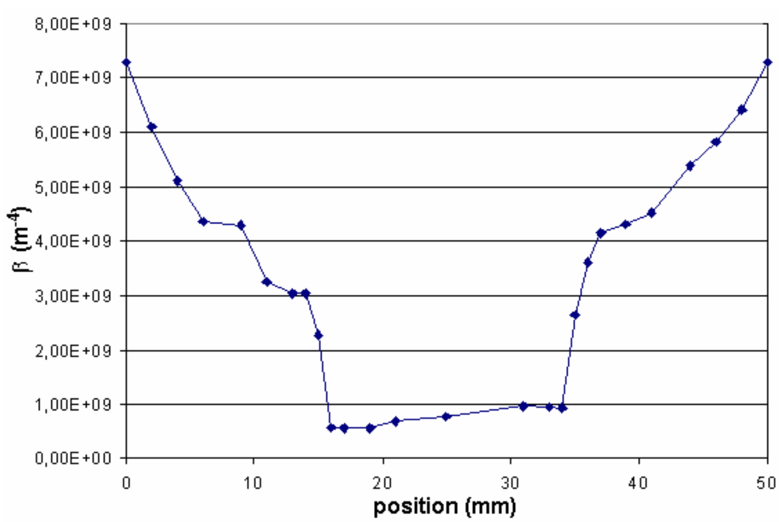


Figure 41 β_{rod} Values Calculated via CFD Simulation, for Different Control Rod positions

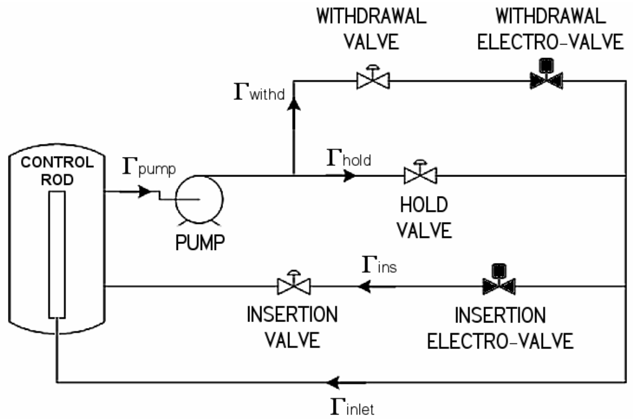


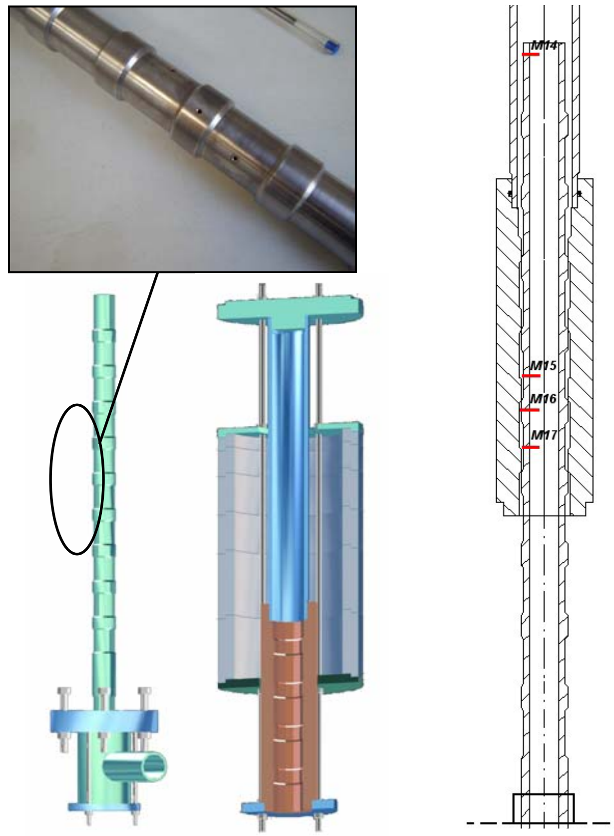
Figure 42 Command and Control Scheme for the Hydraulically Driven Control Rod

In order to obtain a withdrawal step operation, the hydraulic resistance of the Hold Valve has to be reduced, thus allowing the flow rate to reach the head of the piston with increased pressure with respect to the equilibrium value. This can be accomplished by opening a parallel path to the Hold Valve, reducing in that way the whole hydraulic resistance of the circuit ahead of the control rod. A withdrawal electro-valve, normally closed, is opened for a fixed period of time to supply thrust to the piston sufficient to move one notch upward. The hydraulic resistance of the withdrawal parallel path is a design parameter defined by the Withdrawal Valve opening.

Similarly, to obtain an insertion step the pressure drop is reduced in the control rod side: the opening of a parallel path to the control rod reduces the hydraulic resistance in that part of the circuit, causing an increase in the flow rate, a corresponding increase in the Hold Valve ΔP_{hold}, hence a decrease in the $P_{equilibrium}$ value under the piston head. An insertion electro-valve, normally closed, is opened for an assigned time step in order to bypass the drive mechanism. The hydraulic resistance of the path is determined by the Insertion Valve opening.

A preliminary experimental campaign was set up to test the feasibility of the concept and to validate the CFD and the system dynamics analyses. A scaled facility was built, at low (ambient) pressure and temperature, with a test section of reduced length: a 500 mm total length control rod, sufficient to test the labyrinth made up by four periodic profiles at the internal surface of the piston and to simulate withdrawal and insertion transients with at least four successive steps (see Figure 43).

A position measurement device (linear variable differential transformer – LVDT) is mounted on the top of the piston for displacement acquisition during withdrawal and insertion steps movement. The measured data, acquired every 1 mm step in the profile pitch, confirmed the simulated trend and values obtained in CFD simulation (Figure 41). Absolute and differential dp cells are used to measure the equilibrium and thrust pressure on the piston head and pressure drops across the control rod labyrinth, the hold, withdrawal and insertion valves and the pump. A magnetic flow meter measures the flow rate at the control section inlet, while a calibrated orifice is used to measure the flow rate in the inlet path.

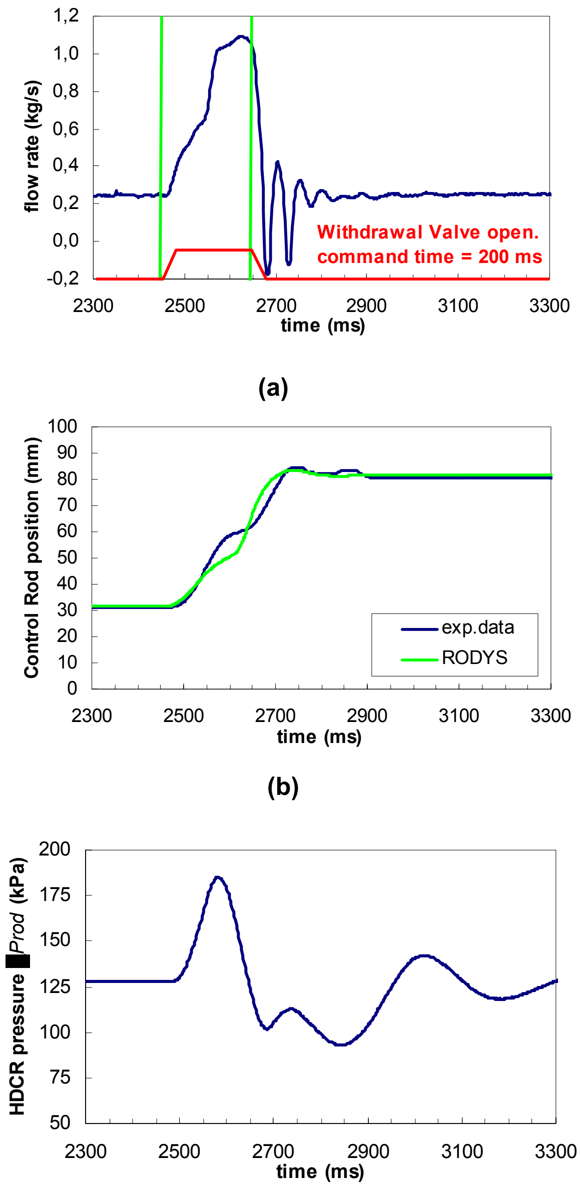


(a) (b) (c)

Figure 43 Hydraulic Drive Test Section: (a) fixed cylinder and a detail showing the pressure tap holes, (b) moving piston, with the four internal profiles and external rings to modify the rod weight, (c) cross section of the assembled piston-cylinder and four pressure tap positions.

First, a dynamic test confirmed the capability of the hydraulic drive device to keep steady state equilibrium positions as expected, without oscillations. Different equilibrium configurations were reached by simply adjusting the Hold Valve opening and the head supplied by the pumps, also accounting for different values of the control rod mass ranging from 12 to 20 kg.

The second part of the preliminary experimental campaign was devoted to the dynamic behavior of the device. Both single and multiple withdrawal and insertion steps were tested, with different control rod weights. Once properly tuned, the control valve opening time periods allowed the transients to be fully repeatable. As an example, a single withdrawal step transient is reported in Figure 44. The inlet line flow rate and the Withdrawal Valve diagram showing the opening and closing instants are displayed in graph (a); the device position showing the 50 mm movement, one notch upward is shown in graph (b), together with the calculated (RODYS) positions with good agreement; the pressure values under the piston of the hydraulic drive, leading to the movement, are reported in graph (c).



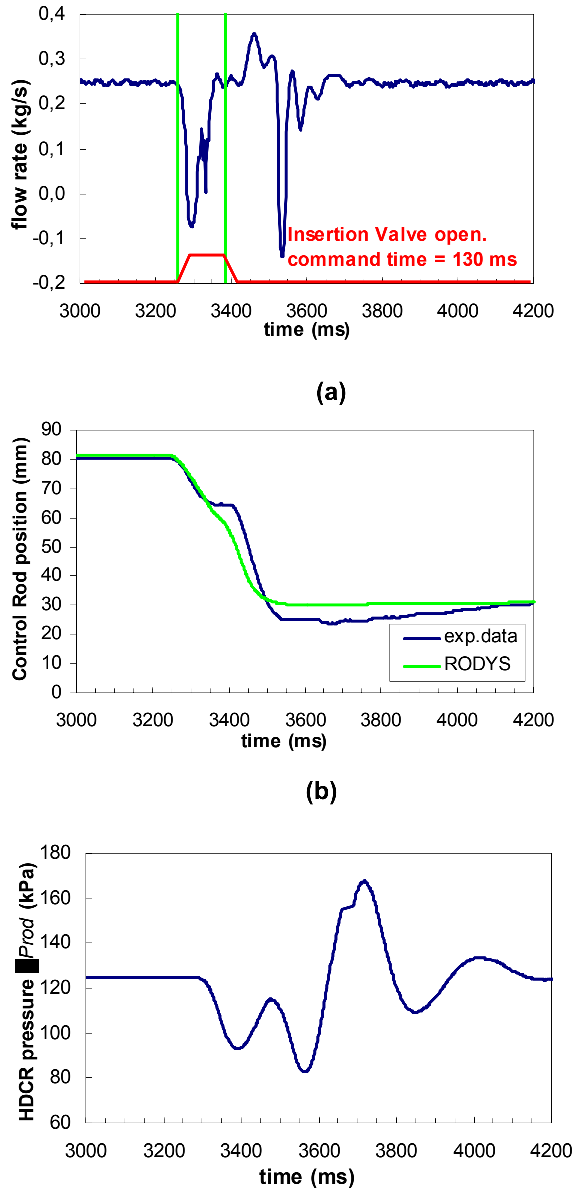


Figure 44 Control Rod Withdrawal Step Measurements

Figure 45 Control Rod Insertion Step Measurements

The same good agreement was found for a single insertion step transient (Figure 45).

A sensitivity analysis was performed to identify the grace-time period for each equilibrium position, i.e. the maximum error on the command time without affecting the control rod movement (see Table 9).

Table 9 Grace-time Periods for Movement of Hydraulically Driven Control Rod

Equilibrium position	Allowed error on command time	
	insertion	*withdrawal*
D	25 ms	35 ms
C	35 ms	45 ms
B	50 ms	60 ms

This analytical and experimentally investigation indicated that the hydraulic drive system is feasible, predictable and stable. Questions still to be addressed are the lifetime behavior in actual operating conditions and the engineering of the drives for each rod in the IRIS core.

8. INTERNAL SHIELDING

As introduced in Section 2, the integral configuration offers the opportunity of dramatically reducing the radiation field on the vessel outer surface with substantial benefits in terms of economics, operation and workers exposure.

These benefits can be summarized by the following four targets:

1. Eliminate the need for RPV surveillance program (required in present PWRs)

2. Provide sufficient gamma shielding to limit the dose outside the vessel from activated internals (barrel, lower support plate) to make it easier and more economical to perform periodic in-service inspections and final decommissioning and disposal

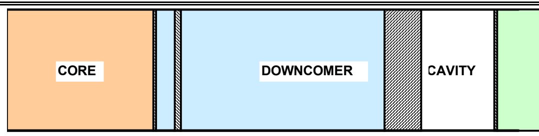
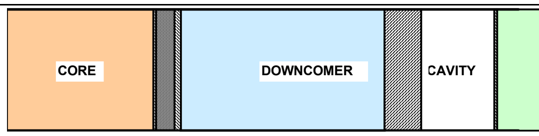
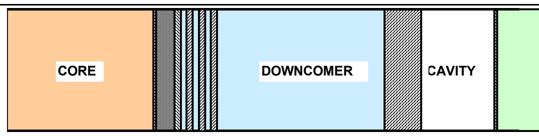
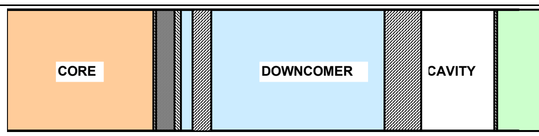
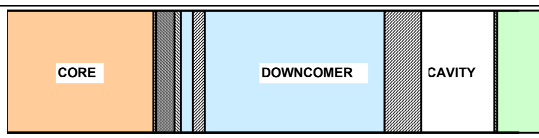
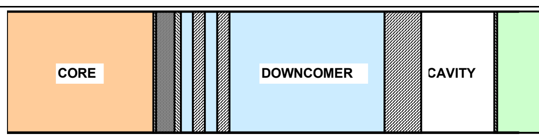
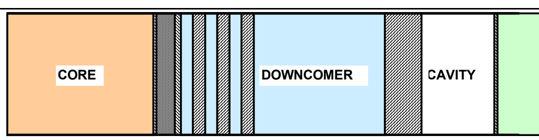
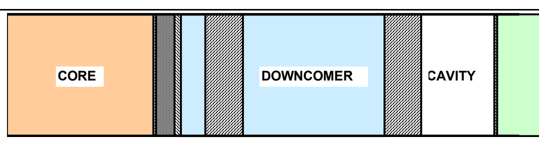
3. Keep cumulative activation of materials outside the vessel (particularly the steel liner and the concrete of the cavity) below the regulatory clearance level, and limit the activation of the vessel itself.

4. Eliminate the need for a biological shield, which will be limited to its structural function.

The first target is the easiest to meet, while the last is the most demanding. The approximately 1 m thick water downcomer provides already a substantial neutron and gammas attenuation, which is furthered by the use of the neutron reflector (Section 6.5). This is sufficient to satisfy target 1. Achieving the more demanding targets 2 through 4 may require additional shielding inside the vessel, e.g., in the form of cylindrical steel plates located between the reflector and the pressure vessel. Thus, extensive analyses were conducted to investigate the effect that additional shielding inserted in the downcomer has on attaining the objectives of limiting the activation of materials outside the vessel, such as the steel liner of the vessel cavity and the capability of using the vessel as a sarcophagus during decommissioning. 1-D and 2-D calculations were performed for many alternative configurations using a variety of Monte Carlo simulations and discrete ordinates modeling at POLIMI and ORNL. The two organizations worked in parallel and independently, thus reducing the calendar time required and allowing checking and validation of the results obtained.

In 1-D calculations the geometry was approximated by concentric cylindrical shells of infinite height, thus neglecting end effects. These evaluations therefore are only accurate in determining the effects of radial neutron attenuation. The results are summarized in Table 10. The reference point for the liner activation is the regulatory clearance limit, which varies from country to country in the 0.1-1 Bq/g range. Thus, the addition of the neutron reflector (whose main purpose is neutron economy, not shielding) alone (case 2) is perfectly adequate in this respect. Addition of the shield plates (e.g., cases 6-8) is however instrumental in further reducing the neutron dose by one order of magnitude and the gamma dose by two orders of magnitude in respect to the reflector only case.

Table 10 Liner Activation and Dose Outside the Vessel (1-D Calculations)

Case	Sketch	Description	Liner Activ. Bq/g	n / γ dose µSv/h (vessel)
1		water reflector, no shielding	0.14	10,000 510,000
2		steel reflector, no shielding	0.05	3,000 300,000
3		steel reflector, 3 plates x5 cm	0.01	700 --
4		steel reflector, 1 plate x15 cm	0.01	700 27,000
5		steel reflector, 1 plate x10 cm	0.02	1,500 --
6		steel reflector, 2 plates x10 cm	0.006	400 --
7		steel reflector, 3 plates x10 cm	0.003	200 --
8		steel reflector, 1 plate x30 cm	0.003	200 1,500

The neutron dose outside the vessel depends primarily on the total shield thickness and not on the distribution of the plates (same doses are estimated for cases 3 and 4, as well as cases 7 and 8). However, the gamma dose is expected to be more sensitive to the actual shield placement.

The effect of radiation streaming was evaluated with 2-D calculations (see Figure 46). The vessel, internals, support skirt, liner and concrete of the cavity are described with axial-symmetric cells, i.e., cylinders of finite height and spherical shell segments. It was found that the outer surface vessel dose above and below the shield plates was almost double what calculated in 1-D and that the hemispherical bottom of the vessel had a more significant activation than the lateral wall. The 1-D calculation also underestimated the activation in the cavity. Table 11 summarizes the 2-D results. Thus, with the incorporation of additional shielding, the maximum liner activation for this configuration is 0.02 Bq/g or less, well below the clearance limit, both for the lateral and the bottom liner.

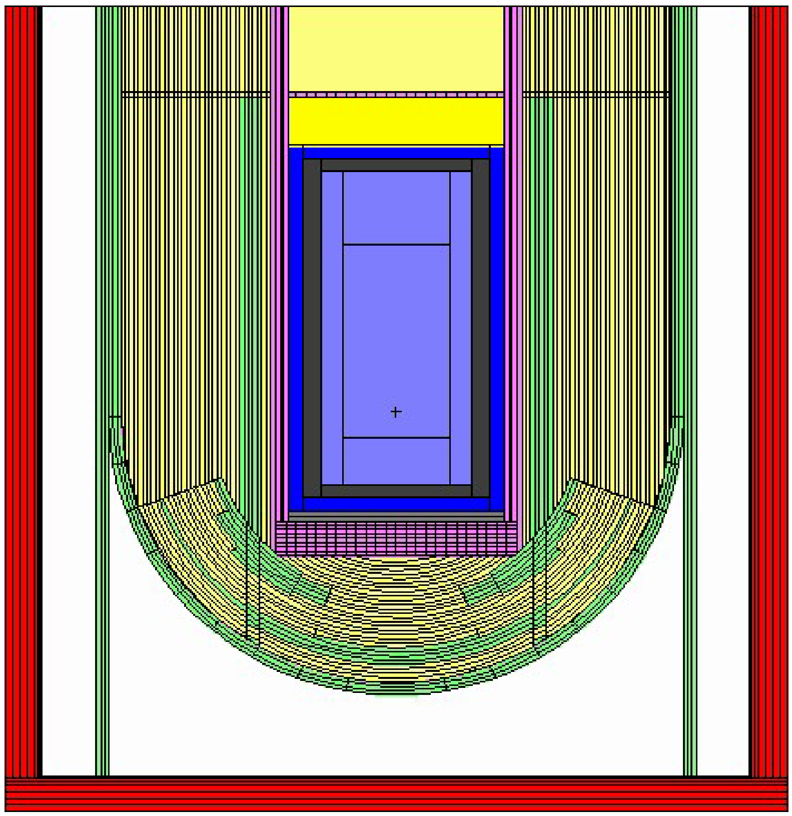


Figure 46 Vertical Section of the Geometry Adopted for 2-D Simulations

Scoping analyses were also conducted to assess the sarcophagus capability of the vessel, with a few results shown in Figures 47a and 47b. In the first case (Figure 47a) of an empty (air filled) reactor vessel, the outer surface dose rate varied from 0.0063 μSv/h near the midplane (point a), to 0.022 μSv/h near the bottom of the downcomer region (point b), to a maximum of 0.1212 μSv/h at point (c) close to the centerline. As expected, these dose rates are all very low. In the second case (Figure 47b) the highly-activated downcomer shield plates are left inside the otherwise empty vessel. In this case, the dose rates outside the vessel were considerably higher, especially on the

Table 11 Outer Vessel Activation (2-D Calculations)

Max neutron flux on outer vessel surface	n cm^{-2} s^{-1}	Where
WITH SHIELDS	2,000	BOTTOM, 70°— 80° (CLOSE TO SKIRT)
WITHOUT SHIELDS	70,000	BOTTOM, 50°— 60°
Max activation of liner, Co 200 ppm	Bq g^{-1}	Where
WITH SHIELDS	0.02	LATERAL, CORE MIDPLANE
WITHOUT SHIELDS	0.6	BOTTOM, CLOSE TO SKIRT
Max activation of concrete, ^{151}Eu 8 ppm	Bq g^{-1}	Where
WITH SHIELDS	0.008	BOTTOM, FLAT DISTRIBUTION
WITHOUT SHIELDS	0.3	BOTTOM, CLOSE TO SKIRT
Max activation of concrete, ^{151}Eu 1 ppm	Bq g^{-1}	Where
WITH SHIELDS	0.001	BOTTOM, FLAT DISTRIBUTION
WITHOUT SHIELDS	0.04	BOTTOM, CLOSE TO SKIRT

lower head, below the downcomer plates. On the outer surface of the vessel, the dose rate was still only 0.032 μSv/h at the midplane (point a), and 0.256 μSv/h near the bottom of the downcomer region (at point b) where the outer downcomer plates still provide significant shielding; but at point (c), on the outer surface of the vessel below the second and third downcomer plates, the dose rate was now about 40.5 μSv/h. However, these calculations were performed without the bottom shielding.

Incorporation of shield plates, especially in the bottom hemisphere will require proper evaluation of the flow distribution to avoid the creation of hot spots. Possibility of local blockages must be excluded; this is more than a theoretical possibility because the plates could act as trap for debris in the areas between the plates and the vessel wall. Addition of the plates will also increase the engineering cost and the weight of the reactor internals.

At this stage of the design, with the specifics of inspections, maintenance, decommissioning and disposal not yet defined, it is impossible to ascertain if the addition of shielding plates is warranted, especially given the fact that even without the additional shielding the integral design IRIS is dramatically better than current LWRs. In fact the IRIS neutron fluence of about 10^{14} n/cm^2 is orders of magnitude less than the current LWR value and even the 10^{17} n/cm^2 threshold which triggers the vessel surveillance program.

The analyses conducted to date have been sufficient to outline the effects of the shielding and its characteristics. Consequently, the project has decided to proceed for now without including additional shielding, ready to later modify the design as necessary.

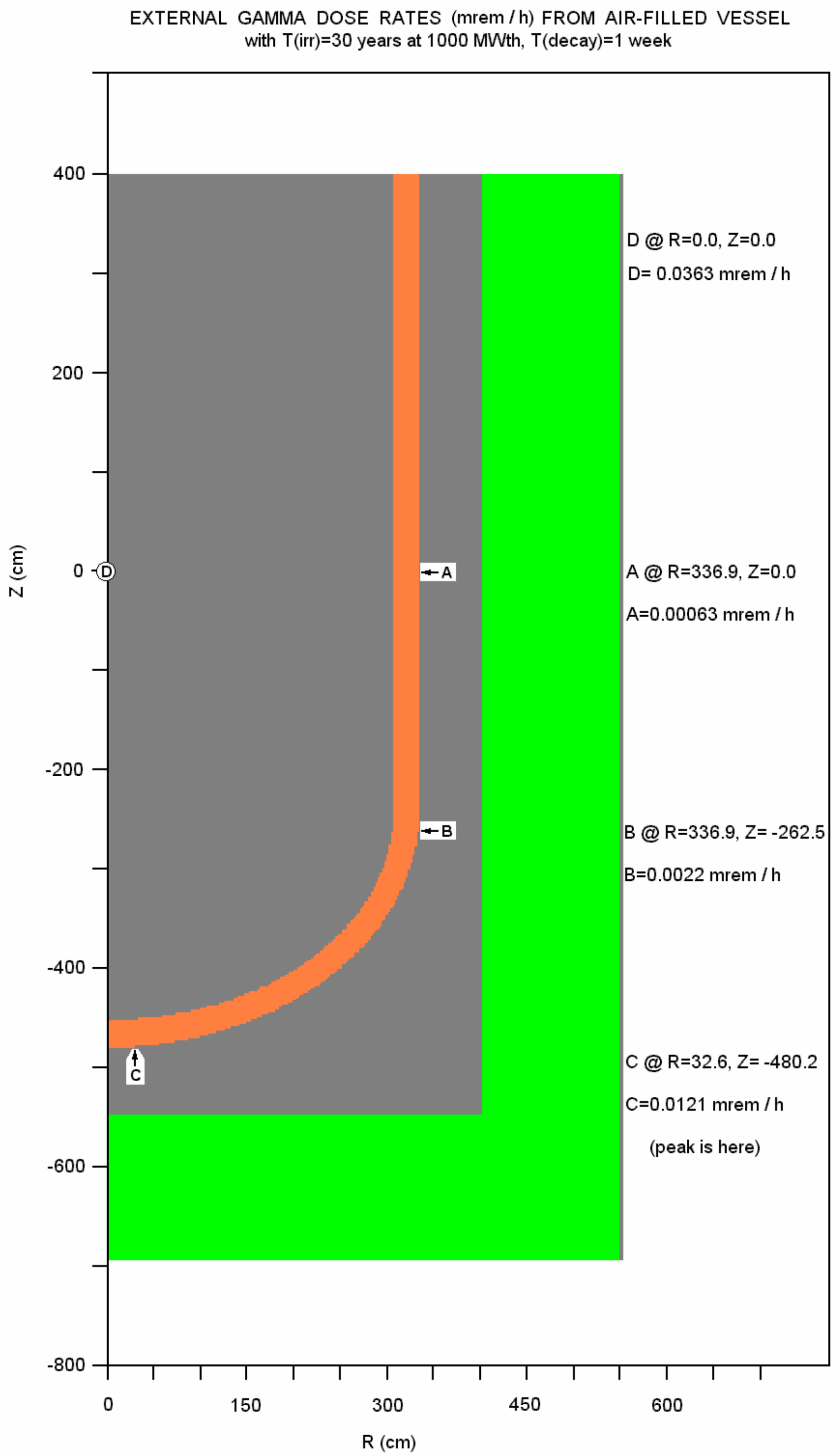


Figure 47a External gamma dose rates from the empty (air-filled) vessel only, based on T(irr) = 30 years and T(decay) = 1 week

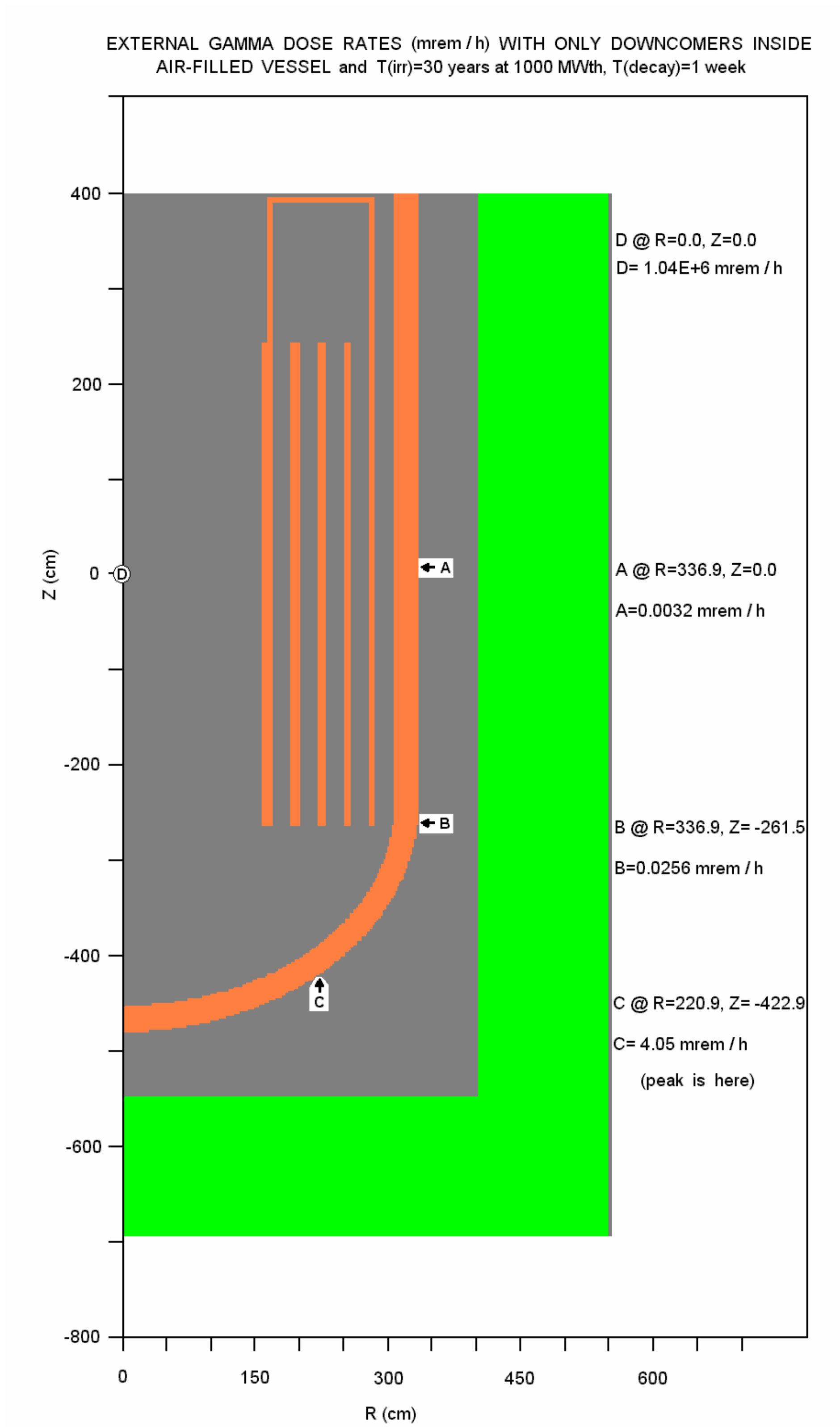


Figure 47b **External gamma dose rates with the downcomers left in the vessel, based on T(irr) = 30 years and T(decay) = 1 week**

9. OPTIMIZED MAINTENANCE

An important factor in improving economic performance is to maximize the time that the plant is on-line generating electricity versus the time spent off-line conducting maintenance and refueling. Maintenance includes planned actions (surveillances) and unplanned actions (corrective maintenance) to respond to component degradation or failure. The term "surveillance" includes a variety of component tests, inspections, overhauls, and preventive maintenance actions.

As discussed in Section 5, a distinguishing characteristic of IRIS is its capability of operating with long cycles. Even though the reference design features a two-batch and a 3 to 3.5 years fuel cycle, selected on the basis of ease of licensing and U.S. utilities preference, IRIS is capable of eventually operating in straight burn with a core lifetime of up to eight years. However, the significant advantages connected with a long refueling period in reducing operation and maintenance (O&M) costs is lost if the reactor still has to be shut down each 18 to 24 months for routine maintenance and inspection. Thus, first and foremost, the IRIS primary system components are designed to have very high reliability to decrease the incidence of equipment failures and reduce the frequency of required inspections or repairs. Next, IRIS has been designed to extend the period between scheduled maintenance outages to at least 48 months. The strategy in extending the IRIS operating cycle length has been "defer if practical, perform on-line when possible, and eliminate by design where necessary". The basis of the design has been a study[21] performed earlier by MIT for an operating PWR to identify required actions for extending the maintenance period from 18 to 48 months. MIT identified 3743 maintenance items, 2537 of them performed off-line and the remaining 1206 on-line. It was also found that 1858 of the off-line items could be extended from 18 to 48 months, while 625 could be recategorized from off-line to on-line. Further, out of the 1858 items there were 1499 electrical surveillances which had a strong potential for also being performed on-line. This left only 54 items which still needed to be performed off-line on a schedule shorter than 48 months. Starting from this MIT study and factoring in the specific IRIS conditions (for example, the 18-month reactor coolant pump lubricating oil maintenance actions performed at PWRs are eliminated in IRIS, since the spool type pumps are lubricated by the reactor coolant), only 7 items were left as obstacles to a 48-month cycle. Four more needed to be performed on line at reduced power. They are reported in Table 12.

Following is a brief summary of the work performed to address the seven identified barriers.

Relief Valves Testing

According to the American Society of Mechanical Engineers (ASME) Code requirements for overpressure protection, [22] as well as relief valve testing requirement given by ASME OMb-2000,[23] all Class 1 Pressure Relief Devices are required to be tested prior to installation, and again within the initial 5 year operating period. Additionally, a minimum of 20% of these valves are to be tested within any 24 months, 50% in 36 months and 75% in 48 months. The routine testing is to determine valve set point, which must be within 3 percent of nominal.

Table 12 Barriers to 48 Month Operating Cycle

4-year cycle, regulatory based – ASME Class 1 (reactor vessel) relief valve testing – ASME Class 2 (component) relief valve testing – Steam generator tube integrity inspections – Safety system operability testing
4-year cycle, investment protection based – Main condenser tube integrity inspections and waterbox cleaning – Main turbine generator throttle control system inspection and cleaning – Main turbine generator trip testing
Reduced power window – Steam and feedwater flow meter calibrations – Steam and feed system large valve maintenance – Auxiliary heat exchanger tube integrity inspections and waterbox cleaning – Auxiliary systems pump and valve maintenance

Of course, the simplest option would be to pursue a code case allowing testing of 100% of the valves every 48 months. Two additional options were evaluated to meet the above requirements, and still allow a 48-month period between maintenance outages; they are:

- Assisted lift devices (such as those by Furmanite) may be used to facilitate on-line testing.

- The use of a Code compliant, isolation valve with appropriate interlocks, to isolate one relief valve (of a redundant pair) for testing.

Either method requires the addition of permanently installed appropriate testing equipment and instrumentation or provisions for access to the relief valves within the containment, and due consideration of personnel safety and working conditions. The IRIS containment is maintained under an inert nitrogen atmosphere, so personnel breathing equipment will be required. However, because of the IRIS inherent shielding, the radiation field in the containment is very low. Either method will also require additional, redundant relief valves.

After an analysis of the pros and cons of the two options, the recommended approach was to use pilot-operated valves, since they are less prone to valve chatter and testing does not require repeated opening and closing of the main valve under flow conditions. Thus, two major damage mechanisms are minimized.

Overall, it is recommended that discussions be first held with ASME on extending to four years the operating period of relief valves. If these discussions prove fruitless, the recommended design option will be pursued using redundant pilot-operated relief valve assemblies with provisions to isolate, test, and/or provide maintenance to the pilot-valve cartridge on a periodic basis. A significant effort may be required to properly design such a system, but designing a system to isolate pilot valves should be a more tractable problem than isolating the main relief valves.

This option will also necessitate the development of the supporting ASME Code Case with provisions and guidelines similar to those shown in ASME Section I code Case #2254. The use of an assisted-lift test device on the pilot valve (or arrangements for test pressurization connections to the isolated pilot valve cartridge) should also be incorporated into this proposed design.

Such a design would accommodate in-situ testing and minimize the effect of valve seat damage resulting from the periodic part-stroke of the main valve. This design configuration could accommodate removal and repair of the pilot actuator, if it becomes necessary. Any such configuration would also be required to meet the requirements of ASME Section III, paragraph NB-7142. This design configuration can provide for periodic in-service verification of the pilot valve's functionality but will need to be coupled with assisted-lift device tests of the pilot valve with it actuating the main relief valve assembly during planned outage periods. Such testing could be conducted at the start of the refueling outage immediately following shutdown of the reactor and prior to the full cool down of the system. In this manner, any identified degradation of the valve could be planned and corrected during the refueling outage.

Steam Generator Tube Integrity Inspections

In-service inspection systems adopted for the Superphénix steam generator were discussed in Section 6.2.2. It was also mentioned that effort is underway under a separate NERI program to develop on-line diagnostic systems.

Safety System Operability Testing

The IRIS Emergency Heat Removal System (EHRS) is discussed in Section 10.2. The steam supply line to each heat exchanger contains a normally open, motor-operated isolation valve, and the heat exchanger return line contains two parallel, normally closed, fail open, air-operated isolation valves and an associated check valve. An EHRS cooling loop is actuated when the isolation valves in the heat exchanger return line to the steam generator feedwater line are opened, and the steam and feedwater isolation valves are closed. This results in steam flow from the steam generator to the EHRS heat exchanger and gravity-driven water flow from the heat exchanger into the steam generator through the feedwater line.

In order to provide the capability to perform frequent periodic testing to demonstrate that the normally closed air-operated valves will open properly while the plant is operating, each of the parallel paths containing the fail open air-operated valve also contains a check valve. Because the pressure in the feedwater line to the steam generators is higher than the pressure in the steam generator steam discharge line, the air-operated valves can be opened, and there is no flow through the EHRS heat exchanger. Thus, these valves can be periodically actuated as needed with no impact on plant operation. In addition, test connections on the heat exchanger discharge line will be provided to permit water injection both upstream and downstream of the check valves to allow verification that the valves are not stuck in the closed position.

In addition to the above testing, it is anticipated that demonstrating operability of the IRIS passive cooling scheme will require initiation of cooling and measurement of both cooling loop flow rate and heat transfer to the heat sink. However, this demonstration would require that the main steam and feedwater isolation valves associated with an EHRS cooling loop be closed and the operation of the heat exchanger would subject the plant to a transient and therefore would affect plant operation. Although this testing could be done with the plant in a reduced power condition, it is deemed prudent to defer this testing until the normally scheduled plant outage. The EHRS flow rate and heat transfer determination does not need to be demonstrated on a frequent basis because these performance parameters are dependent on the physical size and arrangement of the piping and heat exchanger, which do not change with time. Thus, the EHRS flow and heat transfer demonstration would be performed as part of the plant cooldown operation prior to a scheduled refueling. During the cooldown operation, the main stream and feedwater valves can be closed, and the EHRS actuated so that the flow and heat transfer can be demonstrated. This portion of the EHRS strategy is similar to the strategy used for the AP600/AP1000 proof of operability requirement adopted for its single passive residual heat removal heat exchanger cooling loop.

Condenser Cleanliness

Condenser fouling may be due to:

- Macro-fouling: water borne debris are trapped against the tube sheet and they block, entirely or partially, a number of tubes. Split water boxes are commonly used to allow cleaning at partial power.

- Biological fouling: this is perhaps the most important mechanism and is caused by micro organisms. Oxidizing biocides or tube scrubbing devices are used.

- Physical fouling: due to suspended solids deposits, usually controlled through the water velocity.

- Chemical fouling: oxidation and precipitation of dissolved iron or other metals, as well as various corrosion mechanisms. Usually addressed by proper choice of tube materials resistant to chemical fouling.

The following recommendations were made to maintain proper condenser cleanliness in IRIS.

1) A divided water box should be used that will allow access to the condenser at part load. Economic optimization studies are recommended to determine the tradeoffs between 2 divisions and 3 or 4, as well as the merits of over-sizing the condenser.

2) Careful attention should be given to the selection of trash racks or screens. Nominally self-cleaning systems are available and should be evaluated against the estimated labor costs involved in manually maintaining these systems. In any case, plans for final handling and disposal of material removed from the racks or screens should be developed.

3) Sponge ball cleaning systems are recommended to maintain cleanliness in the absence of macro-fouling. Biocide treatment may be considered to enhance control

of microbiological fouling, if local regulations permit. The use of alternative biocides, such as ozone, should also be considered.

4) Cooling water velocity should be maintained high enough to prevent suspended solids deposits, but low enough to avoid erosion of condenser tubes. Generally this will be between 1.8 and 2.4 m/s.

5) Corrosion resistant condenser tubes are recommended, such as stainless steel or titanium.

6) Condenser heat transfer should be monitored to detect fouling problems before they become severe.

Turbine Control Inspection and Cleaning

The Electric Power Research Institute (EPRI) has sponsored several studies of turbine electro-hydraulic controls (EHC). EPRI TR-107069[24] included a review of Licensee Event Reports filed with the U.S. Nuclear Regulatory Commission. In the period from January 1990 through June 1996, about 50 serious events (mostly reactor trips) were associated with the hydraulic portion of the EHC system. These events were further analyzed to determine root causes if possible, and other contributing factors.

One of these factors is fluid contamination and EPRI has devoted a more recent publication[25] entirely to EHC fluid maintenance. It contains recommendations for fluid selection, storage, make-up, sampling and analysis, condition monitoring, purification and troubleshooting.

TVA has experienced no problems with sludge in the EHC fluid of nuclear plants, but it did report sludge for a coal fired plant. Following the EPRI recommendations, especially the periodic monitoring of fluid conditions should allow avoidance of serious problems. Monthly sampling are suggested and in addition the design should permit easy change of all filtration media while on-line.

Turbine Generator Trip Testing

No effort yet has been devoted to this topic, pending selection of the IRIS turbine supplier.

It appears therefore that while significant challenges still lay ahead, a 48-month maintenance interval is indeed feasible for IRIS. The associated O&M cost reduction stems from the increased plant availability and the reduced personnel costs, due to the reduced refueling and maintenance outages. A preliminary assessment is that the IRIS O&M cost should be approximately 20% lower than traditional PWRs.

10. SAFETY

Enhanced safety was one of the requirements set in the NERI program and IRIS has made the quest for enhanced safety the beacon for its development, aided in this by the excellent potential of the integral configuration. However, it has not been a blind quest for safety at all costs. Another NERI requirement was improved economics and the IRIS designers have always kept in mind the truism that the safest reactor is the one which is never built. And, these days if a reactor is not economically competitive, it is not going to be built.

Thus, the IRIS development has been focused on engineering a design capable of providing the highest degree of safety in the simplest and cost effective arrangement. That is, increased IRIS safety must be reached while at the same time costs are reduced. Therefore the design has been based on a three-tier approach:

1. The first tier is the safety-by-design, where accidents are eliminated from occurring or their frequency and/or consequences are lessened. This is accomplished by developing a design that does not require any additional dedicated safety system, either active or passive. And, the IRIS design solutions are to be cheaper than in loop PWRs.

2. The second tier is represented by the simplified passive safety systems which protect against and mitigate the consequences of those accidents not covered or only partially affected by the safety-by-design. The IRIS passive systems are less in number and simpler than the AP600/AP1000 systems.

3. Finally, active systems are adopted to respond to normal and abnormal operating conditions. There are no active safety grade systems in IRIS. However, the much cheaper active non-safety systems play an important role in decreasing the IRIS core damage function.

Thus, the IRIS safety relies on the safety-by-design and the passive systems, which are discussed in the following sections.

10.1 SAFETY-BY-DESIGN

The tenet of the safety-by-design of eliminating the accidents from occurring, or reducing their impact through design rather than engineering features, is of course not new and nothing more than good engineering. However, it takes in IRIS a new meaning and a new life because the unique characteristics of the integral configuration, if properly recognized and investigated, are amazingly conducive to a thorough implementation of the safety-by-design.

The elimination of large break LOCAs, because the large loop piping of conventional PWRs no longer exists, is only the most visible effect of safety-by-design. Many other possibilities exist, which have been thoroughly investigated and incorporated in the IRIS design. Their implementation is summarized in Table 13 and briefly discussed in the following.

The adoption of an integral layout requires the design of a large vessel compared to other PWRs, with a long riser above the core to allow sufficient space for the placement of the steam generators and reactor coolant pumps in the pressure vessel. This

Table 13 Implications of Safety-by-Design Approach

IRIS Design Characteristic	Safety Implication	Accidents Affected
Integral Layout	No large primary piping	• LOCAs
Large, Tall Vessel	Increased water inventory Increased natural circulation Accommodates internal CRDMs	• LOCAs • Decrease in heat removal • Various events • RCCA ejection; eliminate head penetrations
Heat Removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass Effective heat removal by SG/EHRS	• LOCAs • LOCAs • All events for which effective cooldown is required • ATWS
Reduced size, higher design pressure containment	Reduced driving force through primary opening	• LOCAs
Multiple coolant pumps	Decreased importance of single pump failure	• Locked rotor; shaft seizure/break
High design pressure steam generator system	No SG safety valves Primary system cannot over-pressure secondary system Feed/Steam System Piping designed for full RCS pressure reduces piping failure probability	• Steam generator tube rupture • Steam line break • Feed line break
Once Through steam generator	Limited water inventory	• Steam line break • {Feed line break}*
Integral Pressurizer	Large pressurizer volume/reactor power	• Overheating events, including feed line break • ATWS

* Only accident which is potentially affected in a negative way

provides a large coolant inventory in the reactor coolant system, which is a contributor to the IRIS response to small and medium break LOCAs, i.e., to rely on "maintaining water inventory" rather than "providing coolant injection". The unique IRIS response to small/medium LOCAs is a most significant embodiment of the safety-by-design approach and is further outlined here and then discussed in detail in Section 10.3.1. Also, the large coolant inventory provides a large heat sink that acts to effectively mitigate cooldown and heatup events.

The long riser, and the reduced pressure losses in the reactor coolant system, yield an effective natural circulation flow of coolant in the reactor coolant system to remove decay heat from the core. Finally, the tall riser provides sufficient space to accommodate internal control rod drive mechanisms (CRDMs). As discussed in Section 6.6, internal CRDMs will not only eliminate the potential for an RCCA (Rod Control Cluster Assembly) ejection, but also the CRDMs penetrations in the vessel upper head. Thus, the operational concerns associated with boron induced corrosion of the vessel head penetrations are eliminated by design.

Another IRIS specific feature that is used to inherently mitigate the consequences of postulated events is the placement of the steam generators inside the pressure vessel. Coupled with the large primary inventory, this is a fundamental feature to shape the IRIS response to postulated small and medium break LOCAs. The large steam generators heat transfer surface available inside the vessel is used to remove the heat produced by the core during the event, and provides a mean for depressurizing the reactor coolant system by condensing steam inside the vessel, as opposed to loop PWRs which feature a depressurization system that relies on venting reactor coolant steam/mass to reduce pressure. Thus, coolant inventory is maintained. Also, the effective heat removal through the steam generators and the emergency heat removal systems (see Section 10.2) provides effective mitigation for all the events that require safety grade decay heat removal.

The adoption of an integral layout provides an overall reduction in the dimensions of the reactor coolant system, and thus allows to design a compact, higher-design-pressure containment system (because of the spherical shape and reduced dimensions, the same thin shell stress is reached at higher pressure). During the initial phases of a loss of coolant accident, the pressure in the IRIS containment increases early in the accident, and reaches a higher allowable pressure. This higher back-pressure, together with the depressurization inside the vessel discussed above, provides an inherent limitation to the inventory loss from the reactor coolant system, by effectively and quickly zeroing the differential pressure across the break and thus terminating the small/medium LOCA. Thus, the three IRIS design features (large coolant inventory, heat removal inside the vessel and depressurization by the steam generators, higher design pressure containment) all contribute to maintain the core safely covered without the need for any water makeup or injection. It should be noted that a large margin (almost 30%) to the containment design pressure is provided for all design basis accidents, and that the effective reactor coolant system and containment cooling provided by the Emergency Heat Removal System (EHRS) rapidly reduces the pressure in the containment to minimize containment leakage following a postulated LOCA.

Use of internal spool coolant pumps eliminates the pump shaft break accident, since the spool pumps have no shaft, while the adoption of eight pumps makes the consequences of a locked rotor accident most benign.

The IRIS once-through steam generators, with the primary coolant on the shell side, provide a reduced volume of the secondary side, and this allows the IRIS SG's and steam system, up to the isolation valves, to be designed for full reactor coolant system design pressure. This in turn allows the elimination of the steam generator safety valves, since the steam system is protected by the reactor coolant system safety valves, prevents the reactor coolant system from overpressurizing the steam system, and reduces the probability for piping failures since the steam and feed lines are designed for full pressure. These features play an important role in reducing both the probability and the consequences of postulated steam generator tube ruptures. Not only is the potential for failures reduced since the tubes are mostly in compression (primary coolant on the shell side), but also failure propagation is highly improbable since the tube failure mode is a collapse. Additionally, an effective mitigation is provided simply by isolating the faulted steam generator.

Another feature of IRIS once through steam generators is the limited secondary side water inventory. This reduces the consequences of cooldown events, like a steam line break, but on the other hand the limited available inventory in the steam generators hampers mitigation of heatup events, like a feed line break. As pointed out in Table 13, these latter accidents are the only ones negatively affected by the IRIS design. However, the design amply compensates for the limited heat sink provided by the steam generators, through the large thermal inertia in the primary system and the large steam volume in the pressurizer. Also, the rapid loss of mass from the steam generators provides a means for rapid detection of the fault and actuation of the safety features.

As just mentioned, an effective means for mitigating the consequences of heatup events is provided by another design characteristic of the integral layout. A large volume is available in the reactor vessel head for the pressurizer, which is thus designed with a large steam volume, to provide an inherent mitigation to events causing a pressurization of the reactor coolant system (the steam volume to reactor power ratio is five times larger in IRIS than in advanced passive PWRs, see Section 6.4). This not only allows simplification of the design (IRIS does not feature a spray system nor automatic power-operated relief valves), but it also provides an inherent protection against rapid reactor coolant system overpressurization.

The effect that the IRIS safety approach anchored by the safety-by-design has on typical Class IV accidents is shown in Table 14. The results are quite telling: of the eight Class IV accidents, three are eliminated outright and four more have significantly reduced consequences, so that they are downgraded to a lower class. The only remaining Class IV accident is the Design Basis Fuel Handling accident.

Table 14 IRIS Response to PWR Class IV Events

Condition IV Design Basis Events		IRIS Design Characteristic	Results of IRIS Safety-by-Design
1	Large Break LOCA	Integral RV Layout – No loop piping	Eliminated
2	Steam Generator Tube Rupture	High design pressure once-through SGs, piping, and isolation valves	Reduced consequences, simplified mitigation
3	Steam System Piping Failure	High design pressure SGs, piping, and isolation valves. SGs have small water inventory	Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to critical power) consequences
4	Feedwater System Pipe Break	High design pressure SGs, piping, and isolation valves. Integral RV has large primary water heat capacity.	Reduced probability, reduced consequences (no high pressure relief from reactor coolant system)
5	Reactor Coolant Pump Shaft Break	Spool pumps have no shaft	Eliminated
6	Reactor Coolant Pump Seizure	No DNB for failure of 1 out of 8 RCPs	Reduced consequences
7	Spectrum of RCCA ejection accidents	With internal CRDMs there is no ejection driving force	Eliminated
8	Design Basis Fuel Handling Accidents	No IRIS specific design feature	No impact

The safety-by-design thus represents a formidable first step in the Defense in Depth approach.

An early implementation of the potential offered by the safety-by-design in the prevention and management of severe accidents was to consider the application to IRIS of the Core Melt Exclusion Strategy (CMES) developed by the French CEA, who until the end of 2000 was part of the IRIS team.

The Core Melt Exclusion Strategy[26] is an attractive accident management strategy, since the core melt progression and the consequent phenomena threatening the containment integrity are excluded. CEA stated plans at the time were for CMES to replace the core melt management currently adopted in Europe, and lead to improve the plant safety level and possibly public acceptability. CMES is in principle applicable to most future nuclear plants and it is particularly applicable to low power density, medium size reactors such as IRIS.

A Line of Defense method (LOD) was proposed which could be applied as a basis for the safety demonstration. The LODs are any inherent characteristic, equipment or system implemented into the safety related plant architecture as well as any procedure consistent with the General Rules for Plant Operation (e.g. human actions) which accomplish a given safety function.

Two types of LOD are considered:

- The strong lines (called "a") with a probability of failure of the order of 10^{-3}-10^{-4} per year or per demand, and

- The average lines (called "b") with a probability of failure of the order of 10^{-1}-10^{-2} per year or per demand.

As a design goal, accident situations which would lead to large early releases have to be practically eliminated. As stated by the European Technical Safety Organizations (TSO) "when they cannot be considered as physically impossible, design provisions have to be taken to design them out". TSO also stressed that the "practical elimination" of such accident sequences is a matter of judgement: each type of accident sequences has to be separately assessed. Moreover, the "practical elimination cannot be demonstrated by the compliance with a general cut-off probabilistic value". The IRIS Safety-by-Design approach squarely addresses the TSO requirement.

To meet the objective of 10^{-7}/reactor year, per family and per safety function to prevent severe plant conditions, at least 2 "a" LODs should be implemented for Design Basis Conditions. The key condition for the applicability of this rule is the effective independence of the LODs. When implemented, they must fulfill the principle of functional redundancy, i.e., once the upstream LOD fails, the one downstream is still able to achieve the requested function.

Figure 48 shows an example of LOD implementation where LOD4 represents all the inherent characteristic, equipment, system, and procedure implemented to practically achieve the CMES objectives. The safety related functions for the CMES are represented in Figure 49. Knowing that the robustness of the chain is defined by its weakest link, one can stress the fact that the CMES design and assessment shall proceed with an homogeneous process ensuring the final needed quality: each of the different LOD4/i should be at least equivalent to a strong LOD, i.e., their reliability shall be assessed to ensure that the corresponding order of magnitude is consistent with the 10^{-3}-10^{-4} per demand.

The IRIS safety-by-design is a perfect embodiment of the CMES approach. Regarding the LODs related to maintain core integrity, three of them are fully satisfied by the safety-by-design (maintain coolable geometry/core always covered, decay heat removal from both vessel and containment, water inventory and natural circulation capability of integral reactor) to the extent that the fourth one (inject water) is not necessary (gravity driven makeup water is however available).

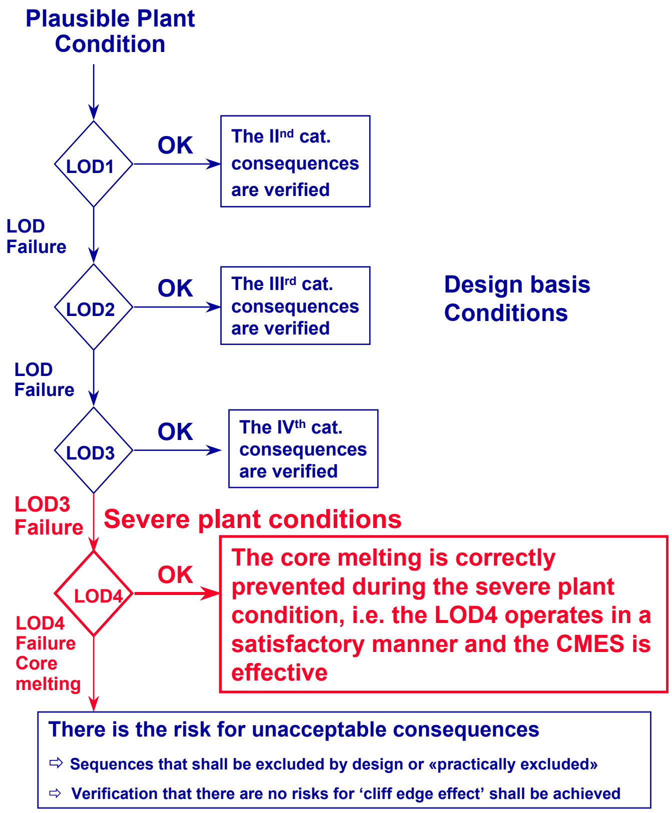


Figure 48 Example of LOD Implementation

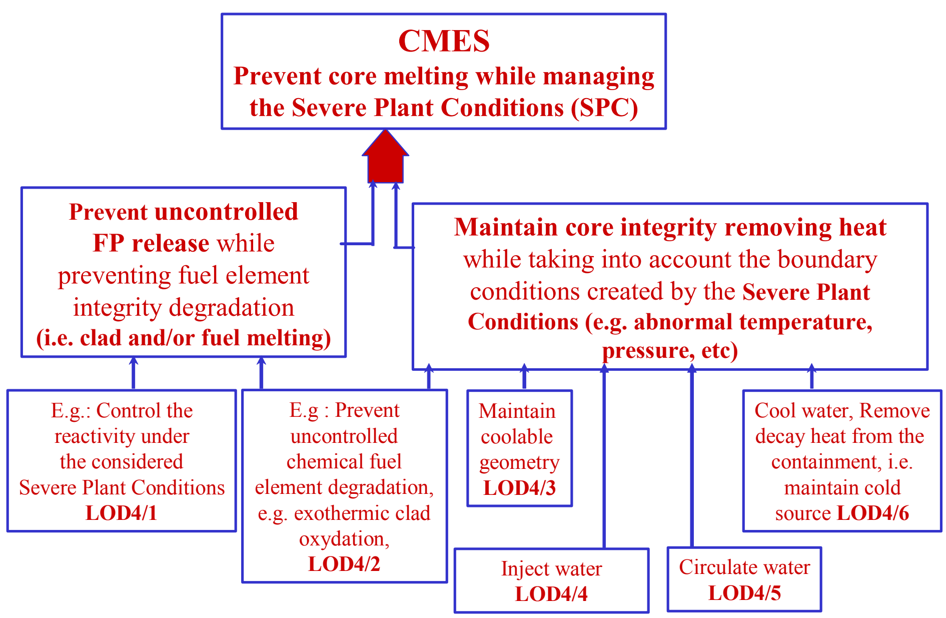


Figure 49 Safety Functions to be Assured Simultaneously to Guarantee the CMES

It is interesting to note that IRIS is the only water cooled reactor which can sustain the simultaneous occurrence of:

1. A LOCA (through safety-by-design)

2. A loss of residual heat removal system (thanks to the vessel-containment coupling IRIS has two independent systems: the SG-EHRS and the air a/o water containment cooling – see Section 10.2)

3. A loss of emergency core cooling (IRIS does not need a ECCS, but gravity makeup water is available anyway).

The capabilities of the integral configuration are not yet widely recognized, as it was reported[27] that while the PBMR can meet the challenge of sustaining the above three simultaneous occurrences, "….you can't assume that sequence for any LWR, even advanced units…"

The superb deterministic safety or defense-in-depth offered by the safety-by-design has of course an important impact on the IRIS licensing, as it will be discussed in Section 13.

10.2 ENGINEERED PASSIVE SAFETY SYSTEMS

The IRIS design builds on the proven technology provided by 40 years of operating PWR experience, and on the established use of passive safety features pioneered by Westinghouse in the NRC certified AP600 and under certification AP1000 plant designs. The use of passive safety systems provides improvements in plant simplification, safety, reliability, and investment protection over conventional plant designs. As in AP600/AP1000, the IRIS passive safety systems require no operator actions to mitigate design basis accidents. Once actuated, these systems rely only on natural forces such as gravity and natural circulation for continued operation. These safety systems do not use any active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as AC power, cooling water, or HVAC). A few simple valves align and automatically actuate the passive safety systems. To provide high reliability, these valves are designed to actuate to their safeguards positions upon loss of power or upon receipt of a safeguards actuation signal wherever possible. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations. The passive systems are designed to meet the single-failure criteria, and probabilistic risk assessment (PRA) techniques are used to verify their reliability.

While the IRIS passive safety systems are patterned after AP600/1000, their number and complexity are significantly reduced thanks to the safety-by-design approach. They are specifically tailored to respond to those remaining accident initiators that are important contributors to the core damage frequency. Thus, the IRIS passive safety systems are even simpler than previous passive safety designs since they contain significantly fewer components, reducing the required tests, inspections, and maintenance; they require no active support systems, and their readiness is easily monitored.

Before outlining the IRIS passive systems it is necessary to present the IRIS containment design because it has a very significant effect in the managing of design accidents, especially small-to-medium LOCAs, as already mentioned in the previous section.

Because the IRIS integral RV configuration eliminates the loop piping and the externally located steam generators, pumps and pressurizer with their individual vessels, the footprint of the patent-pending IRIS containment system is greatly reduced. 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') in diameter (see Figure 50). 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.

Figure 50 shows the pressure suppression pool that limits the containment peak pressure to well below the CV design pressure. The suppression pool water is elevated such that it provides a potential source of elevated gravity driven makeup water to the RV. Also shown is the RV flood-up cavity formed by the containment internal structure which contains the lower 9 meters (~30') of the reactor vessel. This below ground flood-up cavity ensures that the lower section of the RV, where the core is located, is surrounded by water following any postulated accident where coolant mass is lost. 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. The flooded cavity also ensures sufficient heat removal from the external RV surface to prevent vessel failure following beyond design basis scenarios.

The IRIS passive safety systems are shown in Figure 51 and discussed below.

- A passive emergency heat removal system (EHRS) consists of four independent trains, each including a horizontal U-tube heat exchanger located in the refueling water storage tank (RWST) located outside the containment structure that is connected to one of the four separate SG feed/steam lines. The RWST provides the heat sink for the EHRS heat exchangers. The EHRS is sized so that a single train can provide decay heat removal in the case of a loss of secondary system heat removal capability.

  The EHRS operates by natural circulation removing heat from the primary system through the steam generators heat transfer surface. The steam produced in the steam generators (SG) is condensed in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. Following a LOCA where the loss of mass uncovers the SG tubes, the EHRS depressurizes the Reactor Vessel (depressurization without loss of mass) by condensing steam on the SG tubes. Thus, the EHRS contributes to maintaining the coolant inventory in IRIS because it condenses the steam produced by the core directly inside the reactor vessel, while transferring the decay heat to the environment. Also, by depressurizing

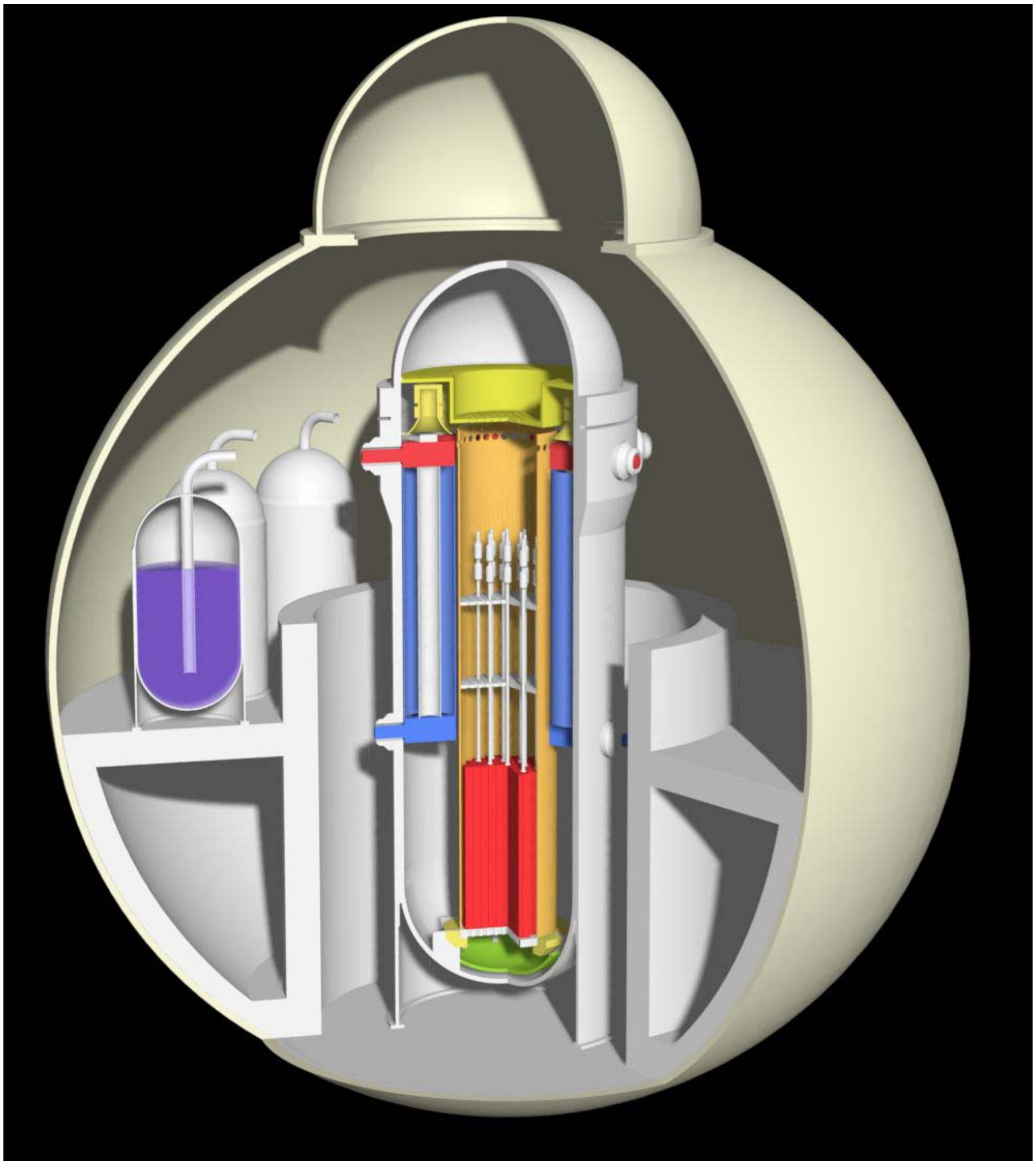


Figure 50 IRIS Containment

the Reactor Vessel, the EHRS limits the break flow and consequently the containment pressurization. The continued heat removal by the EHRS then reduces the pressure of the coupled reactor vessel-containment system. Thus, the EHRS performs the functions of both core cooling and containment depressurization.

- A small automatic depressurization system (ADS), from the pressurizer steam space, assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific setpoint. This ADS has one stage and consist of two parallel 4 inch lines, each with two normally closed valves. The single ADS line downstream of the closed valves discharges into the pressure suppression system pool tanks through a sparger. This ADS function ensures that the reactor

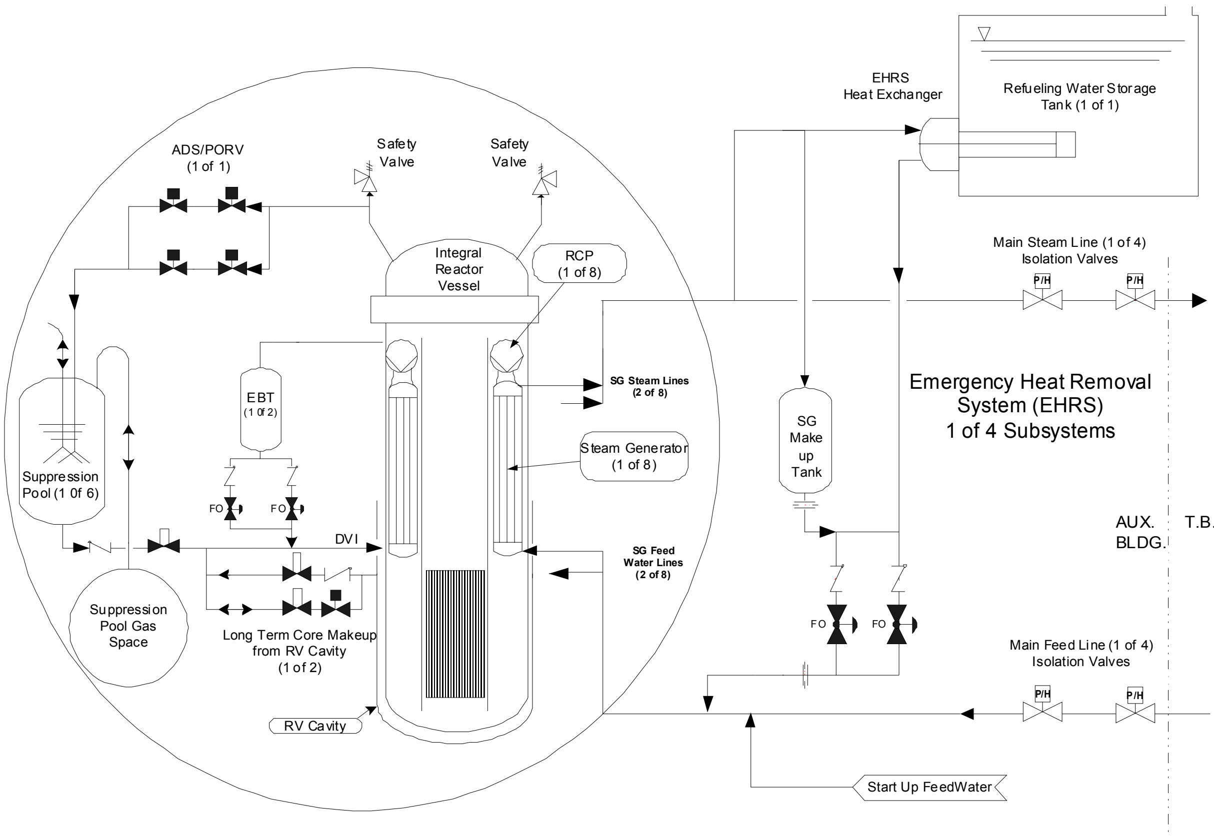


Figure 51 IRIS Engineered Passive Safety Systems

vessel and containment pressures are equalized in a timely manner limiting the loss of coolant and thus preventing core uncovery following any postulated LOCA.

- Two compact (450 ft^3) full-system pressure emergency boration tanks (EBTs) can deliver borated water to the Reactor Vessel through the direct vessel injection (DVI) lines, providing a diverse means of reactor shutdown. By their operation these tanks also provide a limited source of gravity-fed makeup water to the primary system.

- A containment pressure suppression system (CPSS) consists of six water tanks and a common tank for non-condensable gas storage. Each suppression water tank is connected to the containment atmosphere through a vent pipe linked to a submerged sparger to condense steam released in the containment following a loss of coolant or steam/feed line break accident. The suppression system limits the peak containment pressure following a blowdown event to less than the containment design pressure. The suppression system water tanks also provide an elevated source of water that is available for gravity injection into the reactor vessel through the DVI lines in the event of a LOCA.

- A specially constructed lower containment volume collects the liquid break flow, as well as any condensate from the containment, in a cavity where the reactor vessel is located. During a LOCA, the cavity floods above the core level, creating a gravity head of water sufficient to provide gravity driven coolant makeup to the reactor vessel through the DVI lines. The IRIS Long Term Gravity Makeup System (LGMS) also provides a path for gravity injection to the coolant system from the CPSS.

The safety strategy of IRIS provides a diverse means of core shutdown by makeup of borated water from the EBT in addition to the control rods; also the EHRS provides a means of core cooling and heat removal to the environment in the event that normally available active systems are not available. As mentioned in Section 10.1, in the event of a significant loss of primary-side water inventory, the primary line of defense for IRIS is represented by the large coolant inventory in the reactor vessel and the fact that EHRS operation limits the loss of mass, thus maintaining a sufficient inventory in the primary system and guaranteeing that the core will remain covered for all postulated LOCAs. Even though the EBT is capable of providing some water makeup to the primary systems, this is not necessary, since the IRIS strategy relies on “maintaining” coolant inventory, rather than “injecting” makeup water. This strategy is sufficient to ensure that the core remains covered with water for an extended period of time (days) even if no makeup is provided. Thus, IRIS does not require and does not have the high capacity, safety grade, high pressure injection emergency core cooling system (ECCS), characteristic of loop reactors.

Of course, when the reactor vessel is depressurized to near containment pressure, gravity flow from the pressure suppression system water tanks and from the containment will maintain the RV coolant inventory for an unlimited period of time. However, even this function would not be strictly necessary since the core decay heat is removed directly by condensing steam inside the pressure vessel, thus minimizing the amount of primary water leaving the pressure vessel.

The IRIS design also includes a second means of core cooling via containment cooling, since the vessel and containment become thermodynamically coupled once a break occurs. Should cooling via the EHRS be defeated, direct cooling of the containment outer surface is provided and containment pressurization is limited to less than its design pressure. This cooling plus multiple means of providing gravity driven makeup to the core provides a means of preventing core damage and ensuring containment integrity and heat removal to the environment that is diverse from the EHRS operation.

10.3 IRIS RESPONSE TO TRANSIENTS AND POSTULATED ACCIDENTS

The safety-by-design features of the reactor, with their vastly enhanced defense in depth provide an effective means of satisfying regulatory requirements for design basis events. The main effects of this approach on IRIS safety which were presented in Tables 13 and 14, are discussed here in some detail. All the events that are typically studied as part of Section 15 of the Safety Analysis Report, and for which IRIS will present significant differences from current active and passive PWRs, were examined. Analyses were conducted with the RELAP code, which will be presented in Section 10.4.

10.3.1 Loss of Coolant Accidents (LOCAs)

The integral RV eliminates by design the possibility of large break LOCAs, since no large primary system piping is present in the reactor coolant system. Also, the probability and consequences of small break LOCA are lessened because of the drastic reduction in overall piping length, and by limiting the largest primary vessel penetration to a diameter of less than 4 inches. The innovative strategy developed to cope with a postulated small/medium break LOCA by fully exploiting the IRIS design characteristics is discussed in the following.

As previously mentioned, IRIS is designed to limit the loss of coolant from the vessel rather than relying on active or passive systems to inject water into the RV. This is accomplished by taking advantage of the following three features of the design:

1. The initial large coolant inventory in the reactor vessel;

2. The EHRS which removes heat directly from inside the RV thus depressurizing the RV by condensing steam, rather than depressurizing by discharging mass;

3. The compact, small diameter, high design pressure containment that assists in limiting the blowdown from the RV by providing a higher backpressure in the initial stages of the accident and thus rapidly equalizing the vessel and containment pressures.

After the LOCA initiation, the RV depressurizes and loses mass to the CV causing the CV pressure to rise (Blowdown Phase). The mitigation sequence is initiated with the reactor trip and pump trip; the EBTs are actuated to provide boration; the EHRS is actuated to depressurize the primary system by condensing steam on the steam generators (depressurization without loss of mass); and finally the ADS is actuated to assist the EHRS in depressurizing the RV. The containment pressure is limited by the Pressure Suppression System and the reduced break flow due to the EHRS heat removal from the RV.

At the end of the blowdown phase the RV and CV pressure become equal (Pressure Equalization) with a CV pressure peak of approximately 8 bar_g. The break flow stops and the gravity makeup of borated water from the suppression pool becomes available.

The coupled RV/CV system is then depressurized (RV/CV Depressurization Phase) by the EHRS (steam condensation inside the RV exceeds decay heat boiloff). In this phase the break flow reverses since heat is being removed not from the containment, but directly from inside the vessel. Since steam from the containment is condensed inside the reactor vessel and water is provided from the CPSS water tanks, the liquid level in the RV increases. As the CV pressure decreases, a portion of suppression pool water is pushed out through the vents and assists in flooding the vessel cavity. Eventually the RV and CV pressures are reduced below 2 bar_g in less than 12 hours.

The depressurization phase is followed by the Long Term Cooling Phase where the RV and CV pressure is slowly reduced as the core decay heat decreases. During this phase of the accident recovery, gravity makeup of borated water from both suppression pool and RV cavity is available as required. Since decay heat is directly removed from within the vessel and the vessel and containment are thermodynamically coupled, the long term break flow does not depend on the core decay heat, but it is in fact limited to only the containment heat loss.

10.3.2 Steam Generator Tube Rupture

In IRIS, the steam generator tubes are in compression (the higher pressure primary fluid is outside the tubes) and the steam generators headers and tubes are designed for full external reactor pressure. Thus, tube rupture is much less probable and if it does occur there is virtually no chance of tube failure propagation.

Besides reducing the probability of the event occurrence, IRIS also provides by design a very effective mitigation to this event. Since the steam generators feed and steam piping and their isolation valves are designed for full reactor coolant system pressure, a tube rupture event is rapidly terminated by closure of the faulted SG main steam and feed isolation valves upon detection of the failure. Once the isolation valves are closed, the primary water will simply fill and pressurize the faulted steam generator terminating the leak. Given the limited volume of the steam generators and piping, no makeup to the RV is even required; and since the faulted SG is immediately isolated, the release of radioactivity (primary fluid) to the environment will be minimized.

10.3.3 Increase in Heat Removal from the Primary System

Events considered in the preliminary IRIS safety assessment were:

(1) Excessive heat removal due to feedwater system malfunctions

(2) Increase in secondary steam flow,

(3) Steam system piping failure

(4) Inadvertent EHRS actuation

The events in this category present the potential for a reduction in the reactor coolant system temperature. In the presence of a negative moderator reactivity feedback, a decrease in the moderator temperature results in an increase in the nuclear flux, thus the core power. Thus, these transients are attenuated by the thermal capacity of the reactor coolant system. The overpower/overtemperature protection (neutron overpower, overtemperature and overpower ΔT) prevents a power increase that could lead to a departure from nucleate boiling (DNB). In addition, compared to loop-type PWRs, IRIS presents several features that improve the reactor coolant system response to these events and act to minimize the potential for an excessive reduction in reactor coolant temperature. They are:

1. The IRIS once-through steam generators contain a limited inventory of secondary side water, as compared to loop-type PWRs with recirculation steam generators where a large water inventory is available on the secondary side. This large water inventory provides a large heat sink to mitigate heatup events (see next Section 10.3.4), but also creates the cooldown potential for events that increase the heat removal from the primary side. For example in IRIS, following a major rupture in a steamline, the limited SG secondary water inventory will limit the blowdown to the containment, and timely isolation of the feedwater flow will limit the total heat removal from the reactor coolant system.

2. Another design feature that affects the primary system response to these events is the large reactor coolant system water volume and heat capacity. The IRIS heat sink is in fact located in the reactor coolant system rather than in the steam generator. The large inventory in the reactor coolant system mitigates the reduction in the system temperature. The large downcomer also provides a long grace period before any temperature change initiated by the steam generators reaches the core, so that

the system response (reactor trip) will actually occur before the potential for core power increase is realized. The total water inventory in a IRIS steam generators pair is less than 3,500 kg for all operating conditions, which, on a per-MWt basis is between 1/6th and 1/4th of a loop type PWR. On the other hand, the thermal inertia (amount of coolant in the reactor coolant system) that reduces the cooldown rate is 4 to 5 times that of a loop type PWR on a per-MWt basis.

3. The EHRS is designed such that, in case of a spurious actuation, there is no increase in heat removed from the primary system, and thus no potential for a cooldown at power. While the EHRS is capable of removing almost 15% of full power at nominal reactor coolant system conditions, the design is such that the system will only operate following a reactor trip and feed and steam line isolation.

10.3.4 Decrease in Heat Removal by the Secondary Side

Events considered were:

(1) Loss of external load and turbine trip

(2) Loss of non-emergency AC power and loss of normal feedwater

(3) Feedwater system pipe break

The events in this category cause a sudden reduction in the heat transfer rate in the steam generator, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and reactor coolant system pressure rise. The pressurizer safety valves may open to prevent overpressurization of the reactor coolant system. These valves are sized to protect the reactor coolant system against such overpressurization events. Also, assuming the loss of the normal heat sink, the EHRS is actuated to remove decay heat and bring the plant to a safe shutdown condition.

These events are collectively analyzed for the following reasons:

- To confirm that the pressurizer safety valves are adequately sized to prevent overpressurization of the reactor coolant system;

- To form the basis of the required ASME overpressure protection report;

- To ensure that the increase in reactor coolant system temperature does not result in departure from nucleate boiling (DNB) in the core (on a 95% probability/95% confidence limit basis). The Reactor Protection System is designed to automatically terminate any such transient before the DNBR falls below the applicable limit value;

- To verify the capability of the EHRS to remove core decay heat.

Compared to loop-type PWRs, IRIS presents several features that have an impact on the system response to these events, in particular:

1. In conventional PWRs, the steam generator guarantees a large available water inventory and heat sink to remove decay heat before the actuation of engineered safety features become necessary. The IRIS once through steam generators have only a limited secondary water inventory in the tubes and thus a very limited

`intrinsic` capability of removing heat from the primary system when/if feedwater flow is not delivered to the steam generators. Also, since steam flow is rapidly reduced following a loss of normal feedwater to the steam generators, the turbine is rapidly tripped following any loss of feed flow events by closing the fast closure turbine stop valves. This feature (i.e. rapid turbine trip) tends to further reduce the heat removal capability of the steam generators. Following a turbine trip, if the steam dump system is not available, the pressure in the steam system will start to increase, and the heat removed from the primary system by the steam generators will drop rapidly. Once the steam generator pressure reaches the setpoint for the EHRS actuation, the EHRS is actuated to remove decay heat.

2. The integral reactor coolant system provides a large heat sink. Heat-up events resulting from a loss of heat sink (i.e. loss of feed or steam flow) tend therefore to be mitigated by the large coolant inventory available on the primary side, that will reduce the rate of heat-up in the reactor coolant system. Thus ample time for the actuation of the start-up feedwater system or, if it is unavailable, for the actuation of the EHRS, is provided.

3. Another important feature of IRIS is the large pressurizer steam volume available in the upper head. The IRIS pressurizer steam volume is significantly larger than in loop PWRs on a volume per MWth basis. Since the events in this category are typically analyzed to verify that the reactor coolant system pressure remains within the acceptable limits and to verify that no water relief occurs at the pressurizer safety valves, the increased size of the IRIS pressurizer guarantees additional margin and acts to mitigate the response to these events.

The automatic steam dump system of IRIS, together with the reactor control system, is capable of accommodating a full load rejection without reactor trip.

Some results for the first two events (turbine trip and loss of normal feedwater events) are presented here. Results for the third event (feedline rupture) are not reported since the feedline rupture event does not present significant differences from the loss of normal feedwater analyses due to the low importance of the steam generator secondary side water inventory, and due to the fact that the EHRS is sized such that a single EHRS subsystem is sufficient to remove the decay heat. In fact, the rapid depressurization of the steam system leads to a faster actuation of the EHRS, and therefore the feedline rupture events tend to give an even milder primary side transient than the loss of normal feedwater.

Loss of External Load and Turbine Trip

This anticipated transient is analyzed as a turbine trip from full power since this event is more severe, as it results in a more rapid reduction in steam flow, than the total loss of external electrical load, loss of condenser vacuum or other events resulting in a turbine trip.

Due to the rapid closure of the turbine stop valves, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate turbine bypass. Reactor coolant temperature and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control systems are functioning properly. If the condenser is not available, the reactor is tripped and the steam generated is typically relieved to the atmosphere. Additionally, main feedwater flow is lost if the condenser is not available; feedwater flow is maintained by the startup feedwater system to provide adequate residual and decay heat removal capability.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would be expected to trip from the protection and monitoring system if a safety limit is approached. A continued steam load of approximately 5 percent would exist after a total loss of external electrical load because of the steam/power demand from the plant auxiliaries.

If a safety limit is approached, protection is provided by several different trips: low steam/feed flow, high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trips would all be available to mitigate the consequences of the event.

If the steam dump valves fail to open following a large loss of load, the steam generator pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves are sized to protect the reactor coolant system and steam generators against overpressure for all load losses without assuming the operation of the turbine bypass system, pressurizer spray, or automatic rod cluster control assembly control. The pressurizer safety valves can relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure. The pressure in the steam generator system will rise until the setpoint for the actuation of the EHRS is reached. The actuation of the EHRS will provide adequate heat removal capability and rapidly reduce the pressure and temperatures in the reactor coolant system and in the steam generator system. Depending on the initial conditions, the EHRS may actuate before the safety valves setpoint is reached, preventing any release from the reactor coolant system to the containment.

A sequence of events is provided in Table 15 and the pressurizer pressure transient is shown in Figure 52. Conservative assumptions are made in the analyses of this event to maximize the reactor coolant system pressurization. The pressurizer safety valves capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum turbine load. Due to the mild transient evolution in IRIS compared to other PWRs given the large steam volume in the pressurizer, the safety valves are effectively capable of preventing any significant pressure increase beyond their opening setpoint for a complete loss of heat sink event.

The turbine trip is the limiting overpressurization transient, and therefore can be used as an effective example to evaluate IRIS capability to mitigate pressurization events. The base case presented in Figure 52 was developed assuming two or three pressurizer safety valves, sized so that the relief capacity was, on a per-MW basis, identical to typical Westinghouse PWRs. A sensitivity study was performed to verify the impact of reducing the relief capacity of the relief valves, and the results are provided in Figure 53. Not only do these analyses show that a relief capacity equal to 5% of that of current PWRs (on a per-MW power base) is sufficient to maintain the reactor coolant system pressure to within the 110% limit, but even if no credit is taken for the safety valves (relief capacity set at 0) actuation of the EHRS within 100 seconds would be sufficient to prevent overpressurization of the reactor coolant system. As a comparison, it should be noted that if no credit is taken for the safety and relief valves, a typical PWR would reach the 110% limit in 10-15 seconds.

Table 15 Sequence of Event for Turbine Trip

Turbine Trip (2.2.1)

1.	With offsite power available, minimum reactivity feedback, without pressurizer control	Turbine trip, loss of main feedwater flow	0.0
		High pressurizer pressure reactor trip setpoint reached	9.6
		Rods begin to fall into core	11.6
		Initiation of steam release from pressurizer safety valves	19.3
		Peak RCS pressure occurs	19.4
		EHRS actuate	23.9
2.	Without offsite power, minimum reactivity feedback, without pressurizer control	Turbine trip, loss of main feedwater flow	0.0
		Offsite power lost, reactor coolant pumps begin coasting down	3.0
		Reactor coolant pumps undervoltage reactor trip setpoint reached	4.0
		Rods begin to fall into core	4.5
		Initiation of steam release from pressurizer safety valves	N.A.
		EHRS actuate	25.1
		Peak RCS pressure occurs	27.7

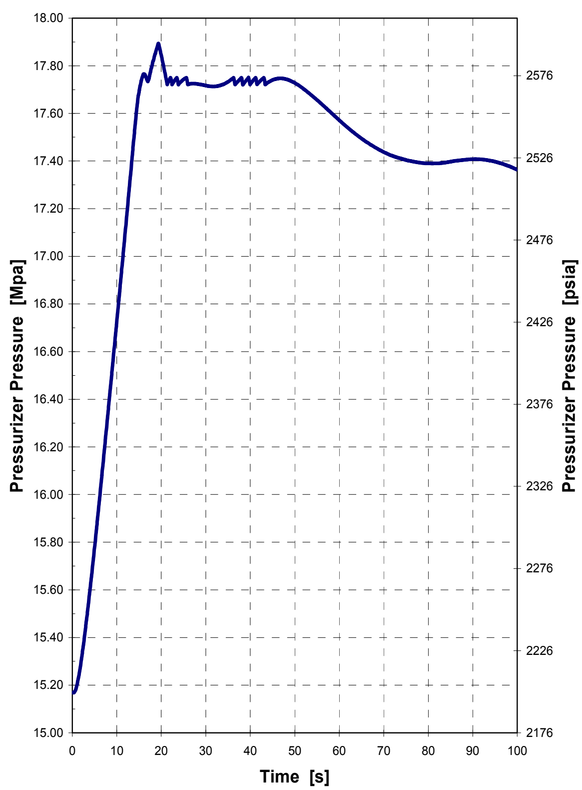


Figure 52 Sequence of Event and Pressurizer Pressure Transient for Turbine Trip with Offsite Power Available

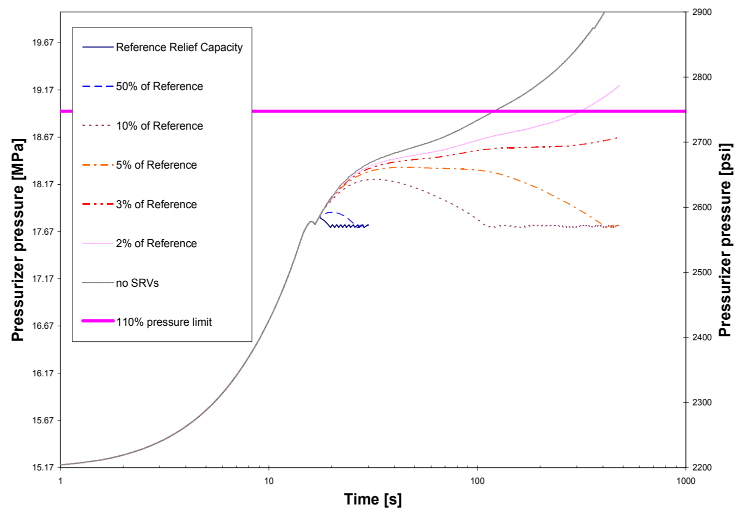


Figure 53 Pressurizer Pressure Transient for Turbine Trip for Different Pressurizer Safety Valves Relief Capacity

Loss of Non-Emergency AC Power and Loss of Normal Feedwater

The loss of offsite power is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby AC power system would typically be available but is not credited to mitigate the accident in these analyses.

From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip because, for this case, the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core.

During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system (which is started automatically when low level occurs in any steam generators pair or when normal feedwater is lost) and by the turbine bypass system. If either of these functions is unavailable, emergency core decay heat removal is provided by the EHRS.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant system and in the EHRS loop.

Following a loss of AC power with turbine and reactor trips, the following sequence of events occurs.

- The loss of AC power leads to a loss of normal feedwater and a loss of forced reactor coolant system flow. The rapid decrease in feedwater flow leads to a reactor and turbine trip on a low steam/feed flow signal. The same signal also actuates the startup feedwater system.

- Plant vital instruments are supplied from the Class 1E and uninterruptible power supply. The onsite standby power system, if available, supplies AC power to the selected plant loads.

- As the steam system pressure rises following the turbine trip, the condenser is assumed not to be available for turbine bypass. As the no-load temperature is approached, the steam dump, if available, is used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition if startup feedwater is available to supply water to the steam generators.

- If steam dump is not available, the pressure in the main steam system will rise until the setpoint for the EHRS actuation is reached. EHRS actuation will isolate the steam generators by closing the main steam and feed isolation valves and will provide decay heat removal in natural circulation through the heat exchanger submerged within the RWST water.

- If startup feedwater is not available, the EHRS is actuated.

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of AC power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related EHRS heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat. The sequence of events following a loss of normal feedwater is very similar to the loss of offsite power sequence, with the main difference given by the fact that offsite power remains available throughout the event.

These events are typically analyzed to verify that the EHRS is adequately designed to remove decay heat when normal means (startup feedwater and turbine bypass) are not available. Therefore, a conservative scenario is assumed and assumptions are made to minimize the heat removal capability of the EHRS. The actuation of the EHRS is delayed due to the low reactor coolant system initial temperatures assumed and due to transient assumptions that tend to delay the actuation of the EHRS on a high steam pressure signal. Note that only the high steam pressure setpoint has been credited for EHRS actuation, and this conservative assumption has been made to demonstrate the large thermal inertia available in the IRIS reactor coolant system that, coupled with the large pressurizer steam volume, provides an extended grace period before decay heat removal becomes necessary. Following actuation of the EHRS, the cooldown progresses until a low temperature signal setpoint is reached. This signal actuates the EBTs.

As the plant cools down, the density of the coolant is increased and this leads to a reduction in the reactor coolant system volume. Therefore, while the cooldown proceeds, assuming that no normal charging and boration is available, the pressurizer volume will empty and the pump suction will be uncovered. In this case a natural circulation path is maintained through the steam generator shroud check valves that open following a loss of forced flow from the reactor coolant pumps. This flow path allows a cooldown of the plant to the safe shutdown condition. The safety valves of the pressurizer may open during the initial part of the transient, but are not actuated during the cooldown since the EHRS heat removal rate always exceeds the decay heat.

A sequence of events and the reactor nuclear power and heat removal at the steam generators during a loss of normal feedwater event are presented in Table 16 and

Figure 54, respectively. Once the EHRS is actuated, it is capable of immediately matching decay heat and a cool-down of the plant proceeds. Figure 55 and 56 show the temperatures in different parts of the reactor coolant system during the event.

Table 16 Sequence of Event for Loss of Normal Feedwater

Accident	Event	Time (seconds)
Loss of Normal Feedwater		
	Feedwater is lost	0.0
	Low Feed Flow Reactor Trip reached	0.1
	Rods begin to fall into core	4.1
	Pressurizer safety valve open (First Time)	383
	EHRS actuation signal reached on high steamline pressure	1657
	Feed and Steam Line Isolation Completed	1672
	EHRS valve completely open	1674
	Maximum pressurizer water level reached	~1700
	Pressurizer safety valve close (Final Closure)	1750
	RCP trip on low pressurizer level	2876
	"S" Signal on Low Tcold reached	4166
	Emergency boration tank valve completely open	4198

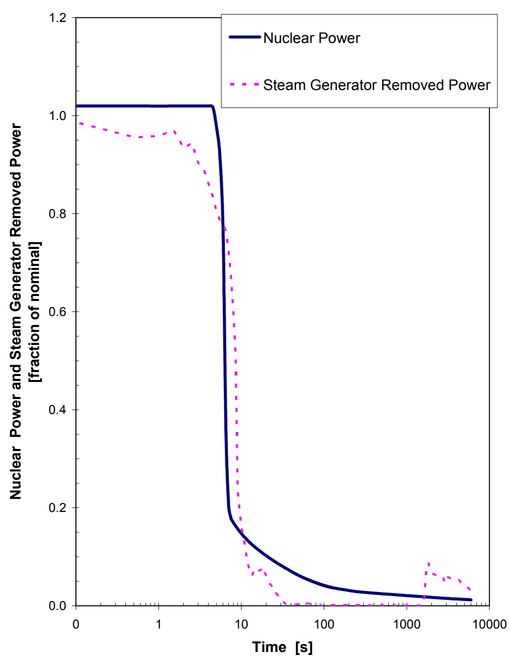


Figure 54 Sequence of Event and Heat Balance for Loss of Normal Feedwater

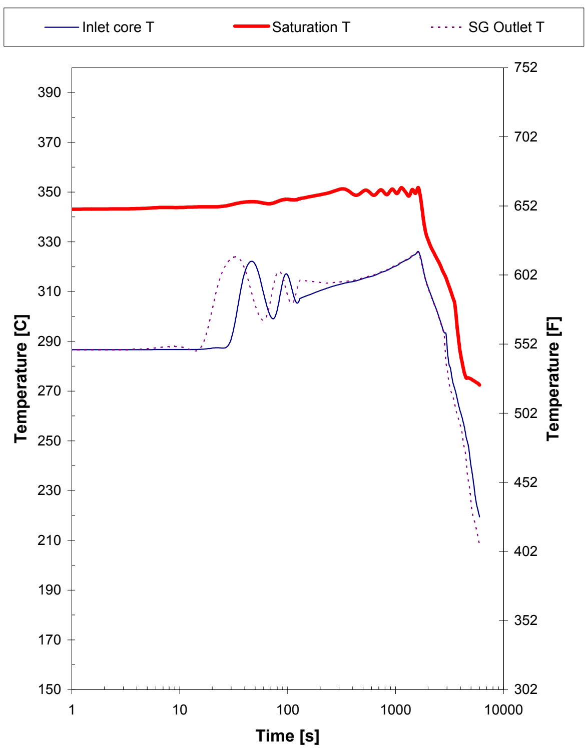


Figure 55 Core Inlet and Steam Generator Outlet Temperature transient for Loss of Normal Feedwater

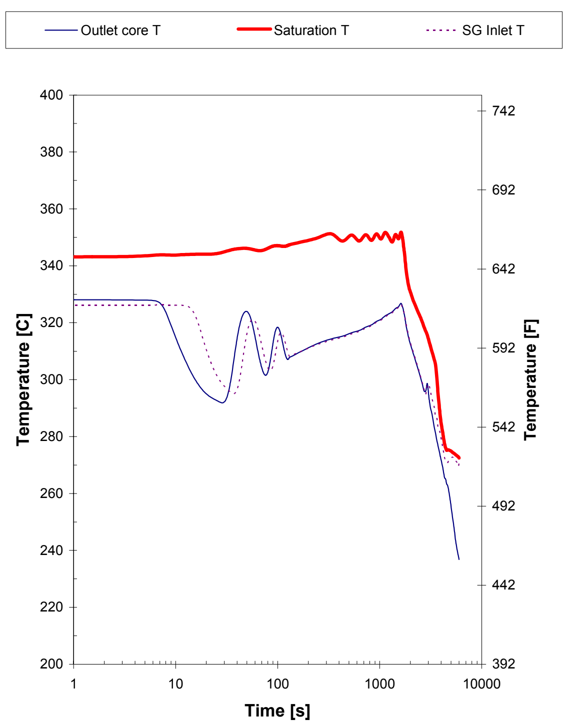


Figure 56 Core Outlet and Steam Generator Inlet Temperature transient for Loss of Normal Feedwater

These results demonstrate that the thermal inertia on the primary side allows a long grace period before actuation of the EHRS becomes necessary to remove decay heat. Once the EHRS is finally actuated, it is capable of rapidly reducing the temperatures in the reactor coolant system.

10.3.5 Decrease in Reactor Coolant System Flow

The following events were considered:

(1) Complete loss of forced reactor coolant flow,

(2) Reactor coolant pump (RCP) shaft seizure (locked rotor)

The events in this category present the potential for a sudden reduction in the heat transfer rate in the core due to the decrease in the reactor coolant flow rate. If the reactor is at power at the time of the accident, the immediate effect of a decrease in coolant flow is a rapid increase in the core coolant temperature. This increase could result in a departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. Protection against these events is provided by a reactor trip before fuel damage can occur.

Complete Loss of Forced Reactor Coolant Flow

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid decrease in the core flow rate, accompanied by an increase in the coolant temperature.

The following signals provide protection for a complete loss of flow accident:

- Reactor coolant pump (RCP) power supply undervoltage (or under-frequency or under-speed)

- Low reactor coolant flow rate

The reactor trip on RCP undervoltage is provided to protect against conditions that can cause a loss of voltage to all RCPs, i.e., loss of offsite power. The reactor trip on RCP under-frequency is provided to trip the reactor for an under-frequency condition, resulting from frequency disturbances on the power grid. The reactor trip on low primary coolant flow is provided to protect against loss of flow conditions that affect only one or some reactor coolant pumps.

The calculated sequence of events for the case analyzed is shown on Table 17, and the calculated minimum departure from nucleate boiling ratio (DNBR), which represents the thermal margin available during the transient, as shown in Figure 57. The RCPs will continue to coast down, and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition is attained and normal plant shutdown may then proceed.

Table 17 Sequence of Event for Complete Loss of Flow

Accident	Event	Time (sec)
Complete loss of forced reactor coolant flow	All operating pumps lose power and begin coasting down; reactor coolant pump undervoltage setpoint reached	1.00
	Rods begin to fall into core	2.50
	Minimum DNBR occurs	4.35

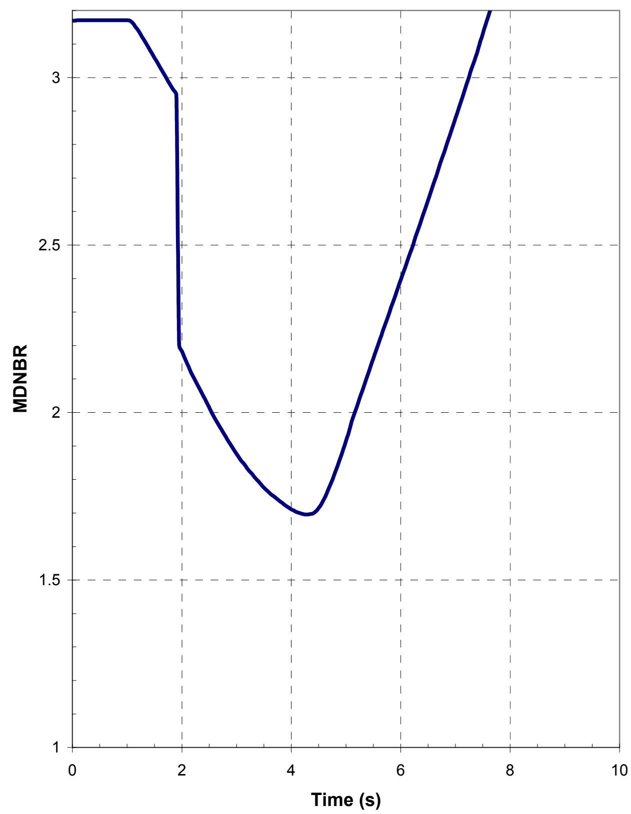


Figure 57 Minimum DNB Ratio Transient for Complete Loss of Flow

Locked Rotor

The accident postulated is an instantaneous seizure of an RCP rotor. Flow through the affected pump and associated steam generator is rapidly reduced, leading to initiation of a reactor trip on a low flow signal.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the tube side of the steam generators is reduced. These two effects combine to result in an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, but it is not expected to lead to an actuation of the pressurizer safety valves. This event is classified as an ANS/ANSI Condition IV incident (a limiting fault) for a loop PWR.

From a phenomenological point of view, the evolution of this event in IRIS does not present significant differences from current loop PWRs, and AP600/AP1000 in particular. However, IRIS has a larger number of reactor coolant pumps (eight versus four for 4-Loop Plants and AP600/AP1000). This leads to a reduced transient following a locked rotor on a single pump. While the phenomenology is similar to other PWRs, the severity of the system response is greatly mitigated by this inherent design feature. The analysis indicates that the more stringent acceptance criteria specified for Condition II events can be met for this event. Additionally, when this feature is coupled with the large thermal margin available in IRIS during normal operation, the analyses performed have shown that the DNB safety limit is not violated following a locked rotor event even if no reactor trip signal is generated.

Table 18 and Figure 58 provide a time sequence and the minimum DNBR transient following a locked rotor event.

Table 18 Sequence of Event for Locked Rotor

Accident	Event	Time (sec)
Reactor coolant pump shaft seizure (locked rotor/broken shaft)	Rotor in one pump locks/breaks	1.00
	Low flow reactor trip setpoint reached	1.20
	Rods begin to fall into core	2.65
	Minimum DNB occurs	3.15
	Loss of offsite power, unaffected reactor coolant pumps begin to coast down	5.34

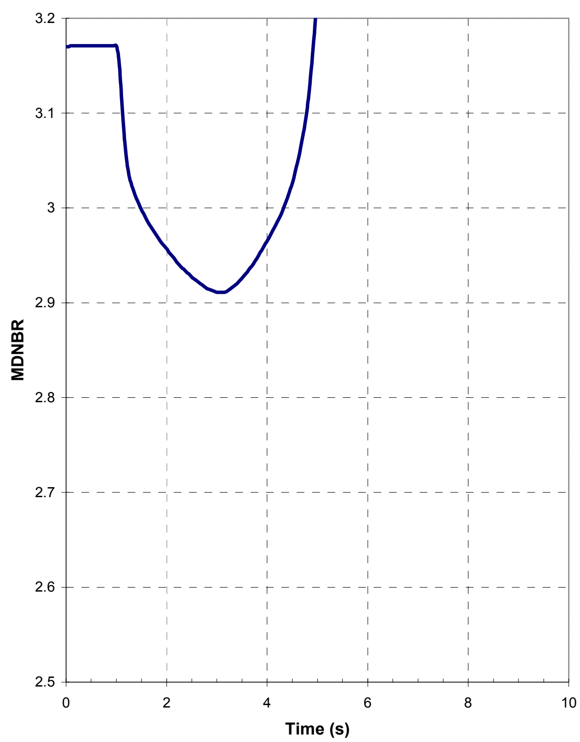


Figure 58 Minimum DNB Ratio Transient for Locked Rotor

10.3.6 Reactivity and Power Distribution Anomalies

These events include:

(1) Uncontrolled rod cluster control rod assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition

(2) Uncontrolled RCCA bank withdrawal at power

(3) RCCA misalignment

(4) Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

(5) Inadvertent loading and operation of a fuel assembly in an improper position

(6) Spectrum of RCCA ejection accidents

No detailed analyses have been performed in this phase for the first five events because the IRIS core is not significantly different from other Westinghouse PWR cores and no major phenomenological differences between IRIS and other PWRs are expected. The sixth event, a Class IV accident, is eliminated in IRIS by the adoption of internal CRDMs (safety-by-design).

10.3.7 Increase in Reactor Coolant Inventory

This category of events is eliminated in IRIS since IRIS does not utilize high pressure coolant injection following a LOCA. The inadvertent actuation of the small emergency boration tanks can be accommodated by the large pressurizer volume with no overpressure or overfill of the RV.

10.3.8 Anticipated Transients without SCRAM (ATWS)

An anticipated transient without scram (ATWS) is an operational occurrence during which an automatic reactor scram is required but fails to occur due to a common mode fault in the reactor protection system or other reason. Under certain circumstances, failure to execute a required scram during an anticipated operational occurrence could transform a relatively minor transient into a more severe accident. As for other Westinghouse plants, ATWS events are not considered to be in the design basis for IRIS.

The improved safety response to several events has resulted in significant reductions in core damage frequency (CDF) and large early release frequency (LERF). However, the contribution to the CDF of some low probability events, such as ATWS, must also be minimized., to support the aggressive IRIS licensing goals (see Section 14.2.2). Therefore, a detailed evaluation of ATWS will be provided as part of the IRIS probabilistic assessment.

10.3.9 Severe Accidents (Beyond Design Basis)

IRIS is designed to provide in-vessel retention of core debris following postulated severe accidents by assuring that the vessel is depressurized, and by cooling the outside vessel surface. The lower part of the vessel is contained within a cavity that always will be flooded following any event that jeopardizes core cooling. Also, like in AP1000, the vessel is covered with stand-off insulation that forms an annular flow path between the insulation and the vessel outer surface. Following an accident, water from the flooded cavity fills the annular space and submerges and cools the bottom head and lower side walls of the vessel.[28] A natural circulation flow path is established, with heated water and steam flowing upwards along the vessel surface, and single-phase water returning downward along the outside of the vessel insulation, to the bottom of the flood-up cavity. AP1000 testing has demonstrated that this natural circulation flow is sufficient to prevent corium melt-through. Application of AP1000 conditions to IRIS is conservative, due to the IRIS much lower core power to vessel surface area ratio. The design features of the containment ensure flooding of the vessel cavity region during accidents and submerging the reactor vessel lower head in water. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. As seen in Section 10.3.1, the IRIS design also includes a provision for draining part of the water in the pressure suppression system water tanks directly into the reactor cavity.

10.4 ANALYTICAL TOOLS DEVELOPMENT AND APPLICATION

Performing analyses at a large number of organizations over four continents can become a technological tower of Babel if attention is not paid to harmonizing the computational tools used. The IRIS project position was to let the various organizations use their analytical tools in the design of those components for which they were responsible and where little interchange of tasks was expected. On the other hand, in the case of those analyses, like transient and safety analyses, where there is a large multi-organizations effort and where NRC approval of the codes used is required, it was decided that validated codes, like RELAP and GOTHIC, should be used throughout the project. Although effort was made not to "develop" codes, still some modifications were necessary because the integral configuration of IRIS is of course different from current LWRs. Discussed here are some of the activities conducted by the consortium regarding the tools used in IRIS transient and safety analyses.

10.4.1 RELAP (Non-LOCA Transients)

The RELAP5/MOD 3.3 code[29] was adopted by all consortium members. The University of Zagreb (FER) was given responsibility for preparation and maintenance of the reference IRIS model and for central coordination of all RELAP analyses. Table 19 provides a breakdown of the various contributors to the transient analyses and their responsibilities.

From a phenomenological point of view, IRIS response to several non-LOCA transients and design basis accidents is not significantly different from other PWRs, so that current Evaluation Models developed by Westinghouse for the analyses of this class of events have been used as a reference in developing preliminary Evaluation Models for IRIS and assessing the areas where challenges might surface. Also, several of the plant systems are based on Westinghouse experience with passive plants and do not pose any new challenge.

Although the RELAP code has been extensively used in the analyses of light water reactors, and has also been used in the transient analyses of the advanced Westinghouse passive plants, the introduction of a new reactor and supporting systems poses great challenges to the development of an appropriate plant nodalization. In

Table 19 Operational Breakdown for RELAP Transient Analyses

IRIS System/Component	Responsible Organization(s)
Primary System and Protection/Control System	
Reactor Pressure Vessel (RPV)	University of Zagreb (FER)
Core Thermal Hydraulic Design	Westinghouse (WEC)
Pressurizer (PRZ)	Brazil Nuclear Energy Commission (CNEN)
Reactor Coolant Pumps (RCP)	Washington Group (W-EMD), University of Pisa (UNIPI), WEC
Steam Generators (SG)	ANSALDO, Polytechnic of Milan (POLIMI)
Reactor Protection System	WEC
Reactor Control System (RCS)	WEC, CNEN, Ansaldo,
Neutronic Feedback Coefficients	WEC
Balance of Plant	
Secondary System	FER
Safety Systems	
Emergency Heat Removal System (EHRS)	POLIMI, WEC
Automatic Depressurization System (ADS)	WEC
Emergency Boration Tank	WEC, FER
Long Term Core Makeup System	WEC
Other Engineered Safety Features (ESF)	WEC

particular, the IRIS integral reactor coolant system layout is sufficiently different from the typical loop PWR to require an ad hoc approach to develop the coolant system model. Based on the best experience acquired by the University of Zagreb in the use of the RELAP code for safety analyses and on Westinghouse experience in PWR analysis, a very detailed plant nodalization was developed. This activity required an extensive effort and was completed during the second quarter of 2002.

The RELAP nodalization used in non-LOCA transient analyses is shown in Figure 59 and is based on the most updated component designs and operational data. While the overall structure is relatively simple and straightforward, the discretization of the components is rather detailed, with 1718 and 1767 volumes and junctions, respectively. A sliced approach has been used in the discretization of the reactor vessel due to the importance of natural circulation in IRIS. Most of the calculational nodes have a linear size in the range of 0.2 to 0.5m. The nodalization was prepared so as to maintain the free volume of the system and elevation differences, as well as core and SG heat exchange areas. The assumption of complete mixing of coolant streams leaving the steam generators was used, with the provision that special mixing models, based on CFD calculations of downcomer and lower plenum, will be introduced in the nodalization later.

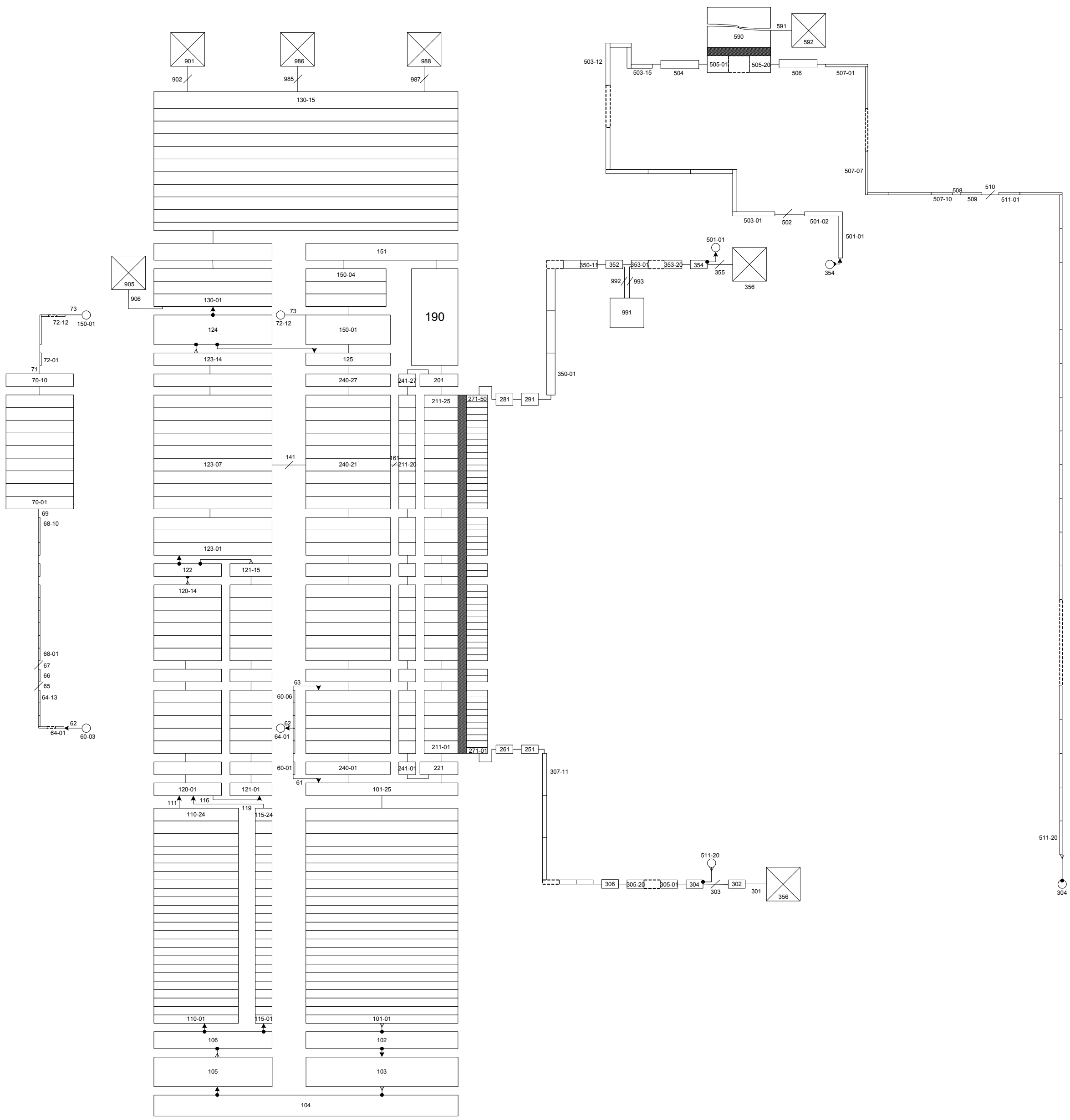


Figure 59 RELAP5/mod3 Nodalization of the IRIS Reactor

The IRIS integral reactor coolant system nodalization is divided in the following main regions:

- Lower downcomer
- Lower plenum
- Core/bypass region

- Riser
- Pressurizer
- Pump Suction Plenum (Upper downcomer)
- Reactor coolant pumps (RCP)
- Primary side of SG modules
- Inactive volume around SG modules
- Inactive volume inside SG modules

Each of the eight RCP/SG modules is explicitly modeled. The possibility of lumping several RCP/SG modules was considered, but it was decided to use an explicit modeling in order to better address physical phenomena, to take into account interaction of SG modules and EHRS loops, to represent the asymmetry due to the different length of feed and steam lines, and to preclude possible recirculation in parallel loops artificially introduced by numerical approximations. The explicit modeling will also facilitate future multi-dimensional treatment and interaction with CFD-like codes.

The pumps are described using preliminary homologous curves in the first quadrant. The pump coastdown when power is lost is described by a table of pump rotation velocity versus time defined according to preliminary design information.

The balance of plant is only partially modeled, and consists of the following regions:

- Feed Lines from the main feed isolation valves (MFIV) to the SG modules
- Secondary side of the SG
- Steam Lines from the SG modules to the main steam isolation valves (MSIV)
- Main feed and steam isolation valves

Two SGs are connected to each feed/steam line. Only one SG is shown in Figure 59, while the actual SG layout is given in Figure 60 for all modules.

Finally, the following Engineered Safeguards Features are modeled: the emergency heat removal system (EHRS), the emergency boration tank (EBT), and the Refueling Water Storage Tank (RWST). These systems are sufficient for the analysis of all IRIS non-LOCA transients and accidents.

Since the remaining IRIS safety features, i.e., automatic depressurization system (ADS), pressure suppression system (PSS), and, the long term core makeup system (LTCMS), establish an interaction between the integral reactor coolant system and the containment, the approach used to model them will depend on the evaluation models used to study LOCA events. A coupling of RELAP (which models the integral reactor coolant system and the secondary side) with GOTHIC (which models the containment) is being developed, as it will be presented in next Section 10.4.2.

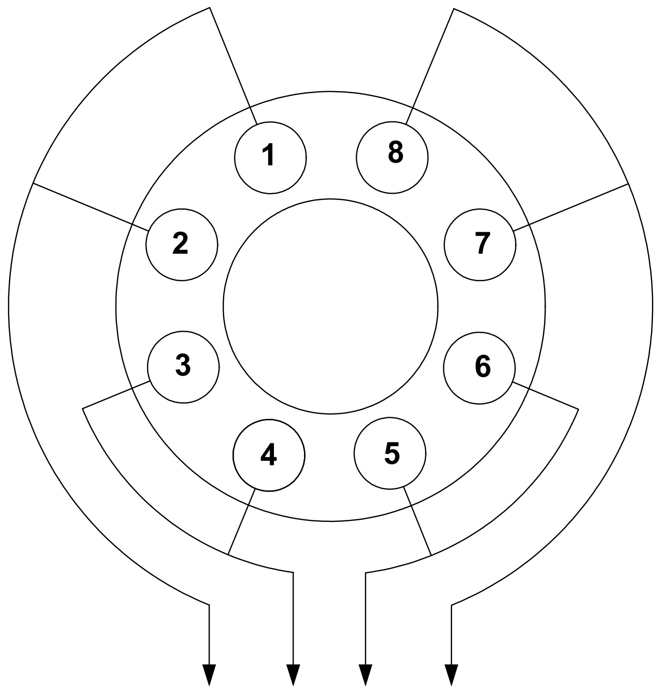


Figure 60 SG Modules Connection Layout

All the main heat structures are included in the model (the total number of heat structures is 1776, with 9553 mesh points). On the primary side the core, baffle, barrel neutron reflector, axial and radial shields, vessel wall, pump casing and some of the internal plates are all modeled to the best detail available. For the SG module the tubes, feed water and steam headers, and the inner and outer shrouds are taken into account. For the steam/feed line and EHRS piping, all the pipe walls are modeled. The outer surface of the reactor vessel is in this phase of the nodalization development assumed to be perfectly insulated. The structures are approximately initialized to the average temperature of their bounding hydraulic volumes.

A preliminary version of the reactor protection system based on the RELAP5 trip model was implemented. The model is continuously improved based on analysis of preliminary accident sequences results. Control variables are provided for calculation of: transferred power (core, SG, EHRS), fluid mass in all relevant parts of the nodalization, and some irreversible pressure losses (current numbers of trips and control variables are 225 and 461, respectively).

The core heat source is based on a power versus time table or point kinetics model. The point kinetics input is preliminary and the kinetics data are calculated for three characteristic burnup cycle points, BOC/MOC/EOC. The total scram reactivity is defined together with corresponding control rod insertion characteristics.

Aside from the development of an appropriate overall nodalization, the main challenges in the development of safety analyses for the integral reactor coolant system are due to the new integral components and to the analyses of mixing effects in the downcomer and pressurizer regions of the system.

The IRIS integral reactor coolant system features three components that present significant differences from other PWRs: the reactor coolant pumps, the steam generators and the pressurizer.

While the IRIS spool pumps are significantly different form both shaft seal pumps (used in current plants) and canned motor pumps (featured in the AP600/AP1000 plants), they don't present significant challenges from the point of view of modeling and analysis using the RELAP code. This is because pumps are defined in the code simply through their hydraulic performance (homologous curves) and coastdown characteristic. Therefore, to correctly model the IRIS spool pumps, it has been sufficient to properly define a sufficient database to represent the pump behavior.

More challenging is the modeling of the helical coil steam generators. Phenomena to be considered are the modeling of flow in a helical tube bundle both externally and especially internally where secondary flow both in single and two-phase inside the helical tubes is caused by centrifugal forces. The potential for parallel channel instability must also be recognized and modeled. Both CFD analyses and an extensive testing campaign will be required to properly characterize the SG thermal-hydraulic characteristics (mainly pressure losses and heat transfer models) over the whole range of conditions over which these components will be operating in normal and abnormal conditions.

Finally, the integral pressurizer does not pose significant challenges from a safety analyses point of view, but it will require a careful design to guarantee appropriate mixing and response to insurge and outsurge events. If the design confirms the preliminary choice of eliminating the spray system, appropriate mixing of the pressurizer water will have to be provided by other means to ensure uniformity in boron concentration with the flowing coolant and detailed CFD analyses may be required. Also, heat transfer and eventually insulation at the boundary between the coolant system and the pressurizer will be studied using CFD analyses.

Mixing phenomena in the downcomer and lower plenum are important in the safety analyses of current PWRs for some asymmetrical events, such as the locked rotor or the steam system piping failures. These asymmetrical conditions are typically studied by defining conservative mixing coefficients that empirically account for mixing and segregation phenomena. Several efforts are devoted to use CFD codes to reduce the conservatism in the analyses of these asymmetrical events.

At first sight, the IRIS integral reactor coolant system layout would appear to increase the importance of 3D and mixing effects, while the proposed nodalization for RELAP safety analyses currently assumes perfect mixing in these regions. This apparent contradiction can be explained on the basis of the very low velocities and long residence times (more than 20 seconds) of the coolant in the downcomer and lower plenum. In fact, in loop-type PWRs mixing effects are typically important for full flow transients, while a more uniform condition tends to exist in low flow, natural circulation conditions. However, this simplification needs to be verified, and eventually appropriate models to represent the fluid mixing in the downcomer region need to be developed. A research program has already been defined. Figure 61 shows a half section of the downcomer and lower plenum that will be used for this analysis. The SG and core conditions calculated with RELAP will be used as boundary conditions for the CFD model, and the core inlet distribution of temperature and flow will be assessed for different operational and abnormal conditions. Temperature, flow and boron mixing phenomena will be

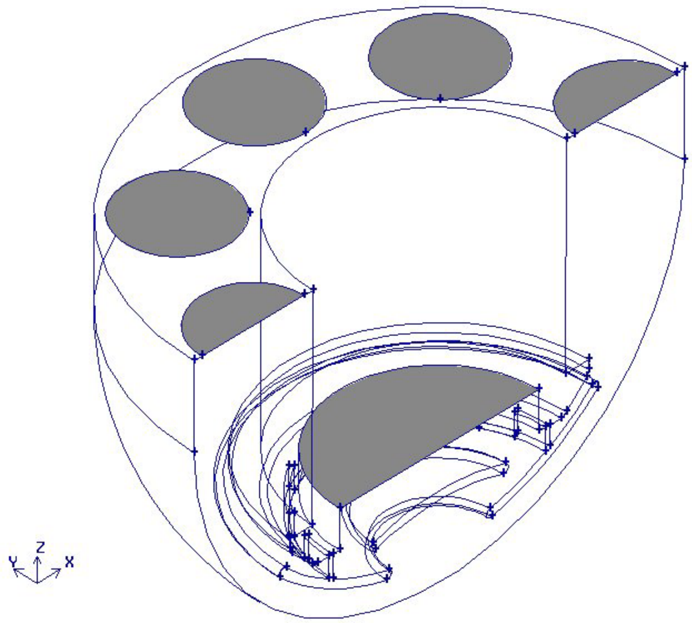


Figure 61 Downcomer 3-D modeling for the CFD code (in dark gray the SG modules outlet and the core inlet, used as input boundary and output boundary for the CFD code, respectively).

evaluated, and the effect of different lower plenum and core support geometries will be assessed.

10.4.2 RELAP/GOTHIC Coupling (LOCA Transients)

As it has been seen in previous sections, the IRIS response to small/medium LOCAs is dictated by the coupling occurring between the vessel and containment following the break. Since RELAP is the code used to analyze the IRIS primary system and since GOTHIC is the code which has been widely used (including by Westinghouse and FER) for containment analyses, the obvious approach to analyze the coupled primary system/containment behavior is to adopt a coupled RELAP/GOTHIC model. Coupling of RELAP and GOTHIC is not new and a rather large experience base exists.[30] For application to IRIS a single direct explicit coupling was chosen, RELAP5/mod3.3, with a version of the GOTHIC code available at University of Zagreb, GOTHIC 3.4e.[31] The connections are at the points of hydraulic contact (the break, ADS, and gravity makeup flow paths). The connections are comprised of a time dependent volume component on the RELAP5 side and a flow boundary condition on the GOTHIC side. The existing detailed RELAP5 model of the reactor coolant system and of the engineered safety features is used for these analyses, together with a simplified GOTHIC model of the containment.

The coupling is direct and doesn't require the use of any additional software tool or protocol. The coupling is explicit in time and RELAP5 is the leading part of the coupled code. Containment conditions from the old time step are used in the RELAP5 new time step system calculation. At the end of each converged RELAP5 calculation time step, interface subroutines transfer the boundary condition data to GOTHIC. GOTHIC then performs one or more time steps and then interface subroutines prepare the containment conditions for next RELAP5 time step, as illustrated in Figure 62.

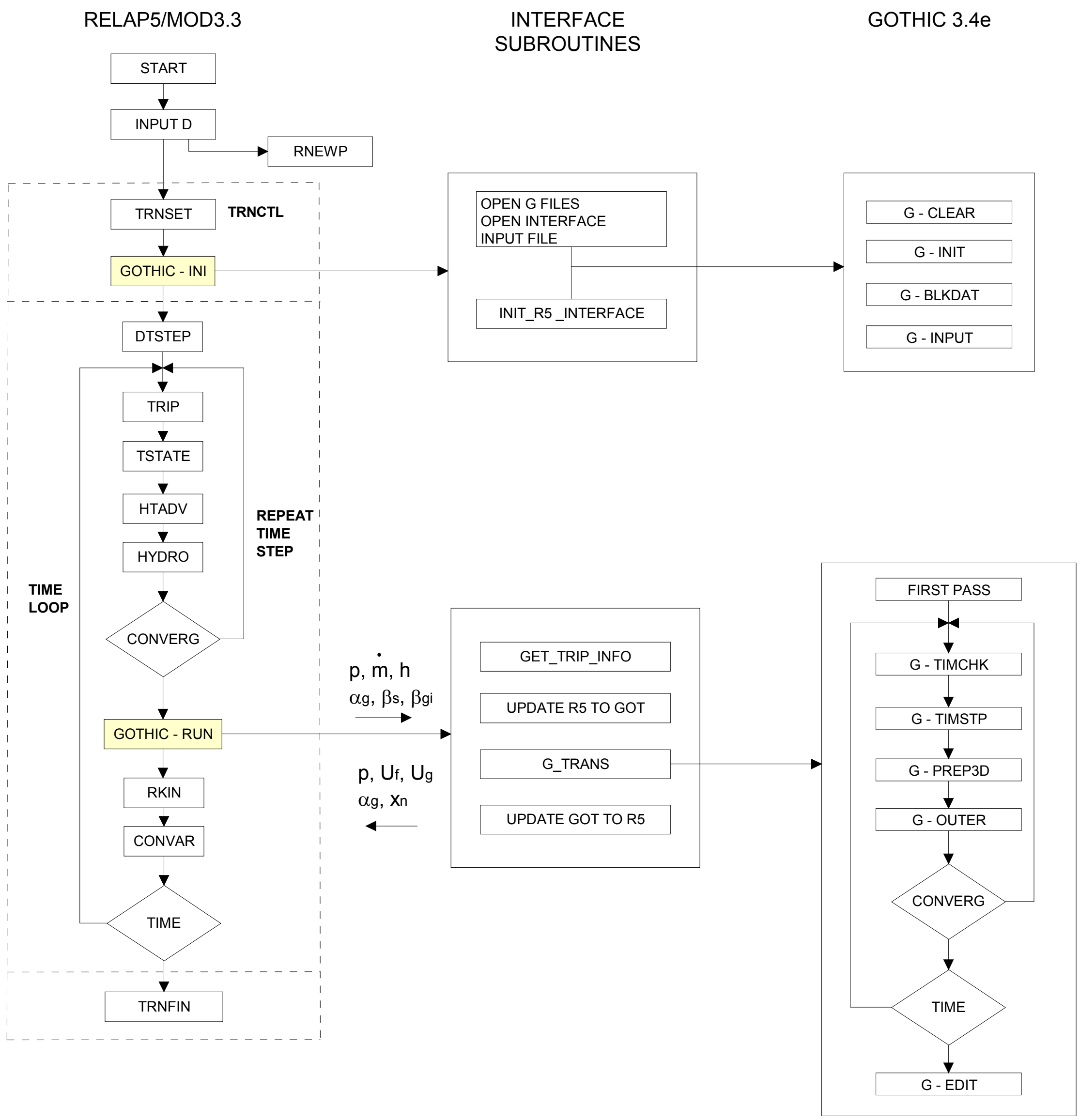


Figure 62 Organization of RELAP5-GOTHIC Coupled Code

The two codes use different main integration variables and the coupling interface has to do the necessary conversions. The interface subroutines responsible for providing GOTHIC data to the RELAP5 code use GOTHIC liquid and droplet data to produce RELAP5 liquid phase data during flow from the containment into the reactor primary system. Conversion of the RELAP5 liquid flow to droplets during blowdown is automatically handled by the GOTHIC flow boundary condition depending on input data.

The variables transferred from GOTHIC to RELAP5 are total pressure, liquid and vapor specific internal energy, vapor void fraction, and non-condensable gas quality. The variables transferred from RELAP5 to GOTHIC are mixture mass flow rate, mixture enthalpy, total pressure, liquid volume fraction, steam pressure ratio, and gas pressure ratios for each of non-condensable gases. The code coupling is inactive during steady state calculations and the original RELAP5 code can be used to produce the initial steady state restart file.

As a result of the chosen coupling scheme, both RELAP5 and GOTHIC are applied to the areas where they can perform best. Multiple connections between the reactor system and the containment models are possible. Connections are not limited to atmospheric regions only; for example, the water level effect on the boundary condition pressure and liquid fraction is taken into account on the GOTHIC side. In addition to coupling the fluid systems it is possible to exchange trip information between the two codes, and the heat structures in one code can be connected to control volumes in another code. GOTHIC's capability to allow subdivision of the containment lumped volumes can be used in the coupled version to perform multidimensional calculations. This will allow proper definition of the other safety systems that need to be modeled in the LOCA analyses of IRIS and will also allow the trip logic for the sequence to be properly defined.

The RELAP nodalization of the primary system was discussed in the previous section. The GOTHIC nodalization of a simplified containment model (SCM) is shown in Figure 63.

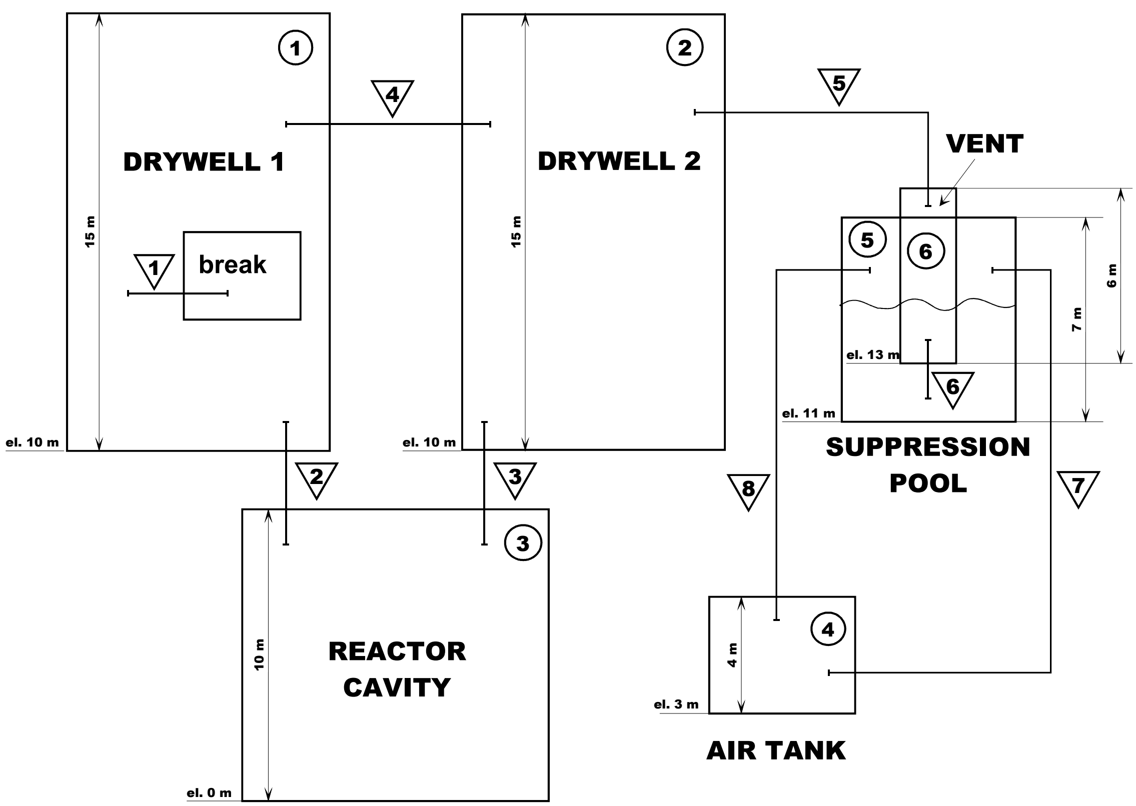


Figure 63 IRIS Simplified Containment Model for GOTHIC

Six control volumes and eight flow paths were used in the SCM model. Heat structures are currently not part of the model. Drywell containment space was split in two parts (Volumes 1 and 2) and connected to the reactor cavity (Volume 3) with two flow paths to simulate mixing between volumes. The same is true for the pressure suppression pool (GOTHIC Volume number 5) which is doubly connected to the air tank (Volume 4). Volume number 6 simulates vent pipes that connect the suppression tanks to the containment atmosphere. The prescribed liquid level in Volumes 5 and 6 determines the initial water inventory in the pressure suppression system. The break position (4-inch break) was assumed to be at the elevation of the reactor coolant pump discharge, which is near the upper portion of the reactor vessel cylindrical section. Another possible characteristic break position (2-inch break) is in the Direct Vessel Injection (DVI) line, just outside the reactor vessel.

The coupling was initially checked by comparison with RELAP5 stand-alone results in a situation where it is assumed that both codes can give similar predictions. The base case for containment modeling (labeled “R5 only” in Figure 64) is one containment node modeled as a RELAP5 branch component (initial conditions at 101.325 kPa, 40°C, nitrogen filled). The corresponding coupled code case (labeled “R5+G”) uses one control volume and one flow boundary condition on the GOTHIC side, and the branch component is replaced with time dependent volume component on the RELAP5 side.

An additional functional test of the coupled code (labeled “R5+G SMC”) was performed for the simplified containment model with six GOTHIC volumes simulating the pressure suppression function of IRIS containment. This case represents a preliminary, more physically true analysis of the small LOCA in IRIS.

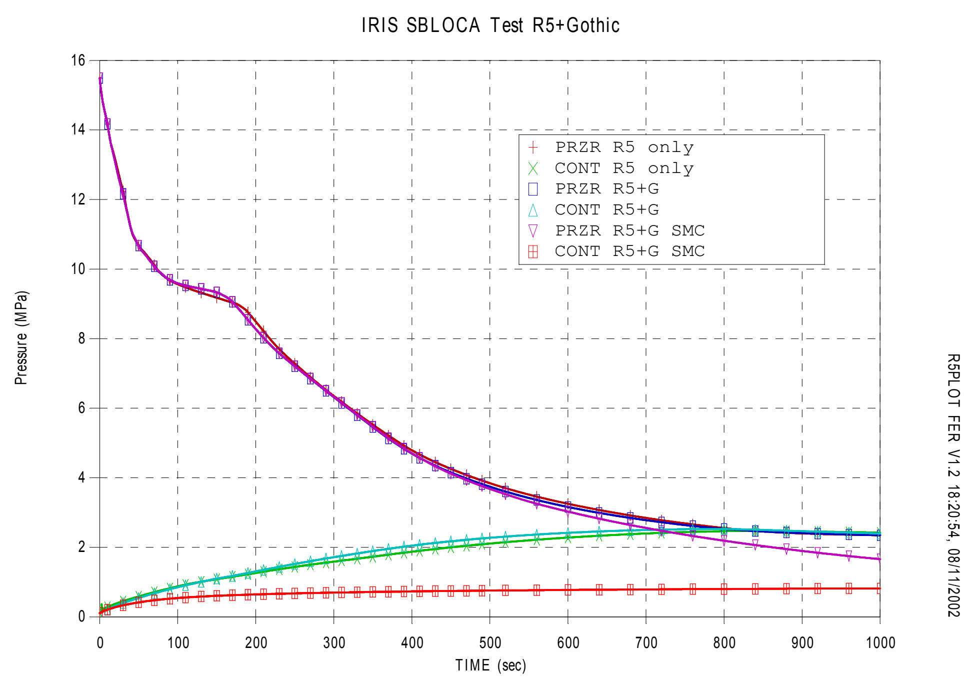


Figure 64 IRIS SBLOCA system and containment pressures calculated by RELAP5/mod3.3 and coupled RELAP5-GOTHIC code for single node and SMC containment

These results of the initial assessment show a good performance of the coupled code, which will therefore be used to perform the IRIS small/medium LOCA analyses, and an appropriate testing campaign will be defined to validate this approach. Preliminary validation will be performed on available test data, for example from the PANDA facility. Still, new testing will be required and will include separate effect tests for new components or operating conditions, and integral effect tests to validate the coupled code.

11. PLANT CONTROL

Reactivity control by a combination of control rods, burnable absorbers and soluble boron has been discussed in Section 5.1. Addressed here is the IRIS project position on plant control. This activity has started only recently, in Spring 2003 and therefore it is still in the formative stage focused on outlining the approach to be followed.

11.1 REQUIREMENTS

The plant control system needs to provide features and characteristic to address not only plant operation at full power but also reactor and plant start-up, power ascension, load-follow, and shutdown operations. In addition, the plant control system should provide the capability for self-validation and adaptation throughout extended operational periods over the plant lifetime. To enhance the economy, efficiency, and reliability of the plant, the IRIS plant control system will make use of integrated control and diagnostic modules to achieve a highly automated intelligent control capability.

The plant control system development approach proposed for IRIS involves determination and verification of control strategies based on: whole-plant simulation; identification of measurement, control, and diagnostic needs; development of an architectural framework in which to integrate an intelligent plant control system; and, design of the necessary control and diagnostic elements for implementation and validation. A key aspect of this development effort is to identify an operational strategy that optimizes plant control while addressing any unique dynamic behavior characteristics resulting from the integral primary system and the once-through helical-coil steam generators (HCSGs). The candidate strategies address coordination of control for pressurizer level, reactor power, and primary coolant average temperature while accounting for the strong coupling between the HCSGs and the primary coolant system in maintaining sufficient secondary coolant inventory. The specification of control, measurement, and diagnostic needs is based on an evaluation of the required command and sensing capabilities, derived from the selected control strategy, to support intelligent control over the full range of operational conditions. To facilitate intelligent control, a supervisory control architectural framework supports the integration of control, diagnostic, and decision elements into a comprehensive, hierarchical command and decision system that can adapt to altered goals or degraded conditions.

The IRIS design, while featuring an innovative engineering of the reactor coolant system, still presents several similarities with loop-type PWRs from the point of view of operation and control. However, some features of the design have an important effect on the plant characteristic response and the design of an intelligent control system for IRIS starts with the identification of the specific features that influence the operational response in various operating modes of the reactor. The most important are summarized in the following.

- Once-Through Steam Generators

IRIS employs once-through steam generators (OTSGs) with helical coils rather than the recirculation SGs used in most PWRs. IRIS steam generators also present a fundamental difference from the OTSGs used in Babcock and Wilcox PWRs: in IRIS secondary water flows inside the tubes and therefore there is no level to measure, consequently no level based control loop can be implemented. Power removed through the OTSGs directly depends on feedwater flow. This means that, following any large loss of main feedwater, the turbine must be rapidly tripped by closing the fast closure admission valves.

Also, IRIS steam generators, with secondary flow inside the tubes and with a 40°C superheating at the steam generator exit, are potentially prone to parallel channel instabilities. Ansaldo tests for the ISIS SG indicated that the instabilities can be avoided through appropriate inlet orificing. Still, IRIS specific tests will be performed and appropriate maps of stable operating conditions (in terms of power, feedwater flow and steam pressure) will be defined to provide input to the protection and control systems. The control function will have to monitor the steam system conditions and provide automatic operational limitations based on these stability maps.

- Large RCS Inventory

IRIS total reactor coolant system water inventory is over 16,000 ft^3, which is significantly larger than any other PWR, especially on a volume-per-MWt basis. This is an important safety feature, since this large heat sink acts to mitigate several events and is a fundamental part of the LOCA response of the reactor. However, this characteristic leads to some differences from current PWRs that impact the design and requirements of the control loop, such as:

– Cooldown/heatup, startup and dilution procedures potentially require more time than in current PWRs. To optimize the plant operations, dedicated heating equipment will be utilized during startup procedures and the Chemical and Volume Control System (CVCS) will be sized to provide sufficient charging and letdown flow for effective management of cooldown/heatup and boron concentration change procedures.

– The low flow velocity coupled with the large inventory leads to a characteristic residence time of about 40 seconds (vs. 10 seconds typical for PWRs). This leads to a system in which the core and steam generators are not as tightly coupled as in current PWRs. In fact, the coolant transit time from the exit of the steam generator to the inlet of the core is approximately 20s. Having an integrated plant control system that can anticipate transients (i.e., a model based control system) can have a significant impact on procedures and lead to better plant utilization.

- Large Pressurizer Steam Volume

As previously reported, the IRIS steam volume at 100% power is about 50 m^3 (>1700 ft^3), significantly larger than any other PWR, especially on a steam volume-per-MWt basis, with the following consequences:

– Improved capability of the system to respond to normal and abnormal occurrences (Condition I and II events), without requiring any safety and relief valve actuation;

– Improved pressurizer response which allows for a design that possibly does not require sprays to reduce pressure increases. This, however, will lead to a slower recovery following transients that will rely on heat losses from the pressurizer. This characteristic needs to be evaluated to confirm whether sprays are required, and how to define the pressurizer heaters control logic to optimize the plant operations.

The functional requirements of the IRIS plant control and instrumentation systems are not significantly different from a loop-type PWR. The plant control system includes the following functions:

1) *Reactor Power Control Loop* – The reactor power control loop coordinates the response of the various reactivity control mechanism; enables daily load follow operation with minimal manual control requirements and is also responsible for axial nuclear power distribution control.

2) *Rod Control Loop* – The rod control loop, in conjunction with the reactor power control system, maintains plant parameters (nuclear power, temperatures, …), without challenges to the protection system, during normal operational transients.

3) *Pressurizer Pressure Control Loop* – The pressurizer pressure control loop maintains or restores the pressure to its nominal value (i.e. within the acceptable deadband) following normal operating transients and avoid challenges to the protection system during normal operational transients.

4) *Pressurizer Water Level Control Loop* – The pressurizer water level control loop establishes, maintains and restores the pressurizer level to its programmed value. The required level is programmed as a function of reactor coolant temperature and nuclear power to minimize charging and letdown requirements. Also, no challenges to the protection system result from normal operational transients.

5) *Feedwater Control Loop* – In a conventional PWR, the feedwater control loop maintains the steam generator water level at a predetermined setpoint during steady state operation; maintains the level within acceptable operating limits during operational transients; and, restores normal water level following a trip. For IRIS, the HCSG level is not a significant process variable, and the feedwater control loop requires a program that uses total steam load as the main process variable.

6) *Steam Dump Control Loop* – The steam dump control loop reacts to prevent a reactor trip following a sudden loss of electrical load and brings the plant to equilibrium no-load conditions.

7) *Rapid Power Reduction* – Several advanced PWRs, such as the AP1000, feature a rapid nuclear cutback (often termed “partial trip”) for large rapid load rejection, to reduce the thermal power to a level that can be handled by the steam dump system. The same function will be provided to IRIS.

8) *Defense-In-Depth Control* – The plant control system provides control of systems performing defense-in-depth functions.

It is evident from the previous list that the principal function of the plant control system is to establish, maintain and restore key process variables to their programmed value (i.e. within the acceptable deadband) following normal operating transients and avoid challenges to the protection system during normal operational transients.

The IRIS plant control system shall perform this principal function during different operating modes (power operation, startup, hot standby, safe shutdown, cold shutdown and refueling) for normal operating transients (step and ramp loads changes, load follow operation, grid frequency response, etc.) and within the permissible deviations defined in the plant Technical Specifications.

11.2 PROPOSED APPROACH

Several of the control loops discussed in the previous section will not present significant difference from current PWR practice. However, the design and specifications of some loops within the IRIS plant control system will differ from current practice, as follows:

- Pressurizer Pressure and Level Control Loop

Due to the large volume available in the upper head region for the pressurizer, IRIS pressure and level control functions will rely more on the inherent response of the design rather than on the actuation of dedicated systems. Pressurizer sprays (with the exception of auxiliary sprays for use during shutdown operation) are currently not included in the design, and no automatic function of power-operated relief valves shall be provided. The system will rely on its large steam volume to mitigate pressure transients. The lack of a spray function will delay the restoration of pressure to initial conditions following some transients, since the system will rely on heat losses to restore the initial pressure. The level control loop will not present significant differences from current practice, and IRIS will make use of the large pressurizer volume to limit requirements on the charging and letdown system following reactor trip and large power reductions.

- Reactor Power and Rod Control Loop

IRIS rod control function will not present significant deviations from current PWR practice. However, due to the large water inventory in the reactor coolant system, IRIS will respond to most of the operational transients (ramp and step load changes) through the rod control loop, to minimize the requirements on the charging and letdown system, essentially making unnecessary (or at least significantly reducing) changes in the boric acid concentration in the reactor coolant system.

- Feedwater Control Loop

The feedwater control loop will present significant differences from current PWRs, especially those plants with recirculation steam generators. This control loop will be more similar to the Integrated Control System of B&W plants and will rely on a more integrated control strategy than for other PWRs. A sliding Tavg program versus turbine load program will constitute the main basis for IRIS control strategy, with a feedwater program based on total steam load in the power range operation. Specific solutions will be implemented for the low and no-load power range to provide a stable plant operation in these regions.

The plant control system design for IRIS will build on recent advances in control theory. Implementation of methods developed in a parallel NERI program[32] will capture the design requirements inside a control engine during the design phase. This control engine then will be not only capable of automatically designing the initial implementation of the control system, but it also can confirm that the original design requirements are still met during the life of the plant as conditions change.

The control engine captures the high-level requirements and stress factors that the control system must survive (e.g. a list of transients, or a requirement to withstand a single failure) and is able to subsequently generate the control-system algorithms and parameters that optimize a design goal and satisfy all requirements. As conditions change during the life of the plant (e.g. component degradation, or subsystem failures) the control engine automatically “flags” that a requirement is not satisfied, and it can even provide recommendations for a modified configuration that would satisfy it.

The implementation of this control-engine design methodology requires the following steps:

1. Determination of Design Requirements Related to Control System Performance

2. Representation of Requirements in Mathematical Form

3. Access to (or development of) a Control Algorithm Library

4. Development and Validation of Plant Models

5. Automated Control Design Generation

6. Evaluation of Control Architectures

7. Control Design Implementation

8. Implementation (or development) of Diagnostics Methods to Update the Plant Model

Since changes to the plant over its long lifetime are slow in nature, it is not envisioned that the control engine would function in a closed loop by automatically changing control parameters or strategies. Its function would be more of an advisory nature through generation of an alert when the original control-system performance requirements are not satisfied under the present conditions (e.g., hardware failures or plant reconfiguration). In addition to the alert, the control engine can also suggest new control system settings that would satisfy the performance requirements under the present plant condition.

IRIS will have a limited operational staff. The combined factors of a reduced operating staff and more complex dynamics means a different approach is needed for overall control of the plant. The solution is to develop a supervisory control system. The role of the supervisory control system is to act as an extension of the human operator to assure safe, reliable operation of the plant. The supervisory control system provides the framework for integrating algorithm-based controllers and diagnostics at the subsystem level with command and decision modules at higher levels that assume increased responsibility while accommodating the human operator’s analytical approach and need to be cognizant of the state of the plant.

The supervisory control structure envisioned for IRIS is hierarchical with a recursive nature. Each node in the hierarchy (except for the terminal nodes at the base) is a supervisory module. The supervisory control module at each level responds to goals and directions set in modules above it within the hierarchy and to data and information presented from modules below it within the hierarchy. Each module makes decisions appropriate for its level in the hierarchy and passes the decision and necessary supporting information to the modules above.

The human operator has the opportunity to interact and direct the goals and actions of the supervisory controller. This interaction may take place directly with any module in the hierarchy. This assures that the human operator can assume ultimate responsibility for the safety and operation of the plant. In addition to the communications up and down the hierarchy, the supervisory controller must keep the operator informed about the status of the plant.

The self-validating controller structure can be easily implemented at higher levels of the supervisory control architecture. By building in appropriate diagnostics, the supervisory control system can determine when subsystem performance has degraded to the point of possibly violating design goals. Once the degradation has been diagnosed, corrective action can be taken by the supervisory control system and the operator can be alerted.

12. POWER PLANT CONFIGURATION

Following inclusion of IRIS in the Early Site Permit (ESP) program, a substantial effort was performed to respond to the three utilities (Dominion, Entergy, Exelon) request for information.

The second year report documented in detail the characteristics of two alternative plant configurations: one comprised of three independent units (1005 MWe total) and a two twin units configuration (1340 MWe total) where each twin unit maximized the sharing of components. Also reported were site related information such as cooling water use requirements, routine emissions and doses, projected release from postulated operational occurrences and accidents, design bases for natural phenomena and labor force requirements. In the third year some additional minor refinements were generated, as requested by the utilities.

Currently work is planned to refine the design of the twin unit arrangement (shown in Figure 65). The twin unit is the preferred configuration since the capital cost can be significantly reduced by the sharing of components; optimization of such sharing is planned. The twin unit, single or in parks, is the appropriate solution for markets like the USA, that require a substantial amount of power installed at once. Smaller power

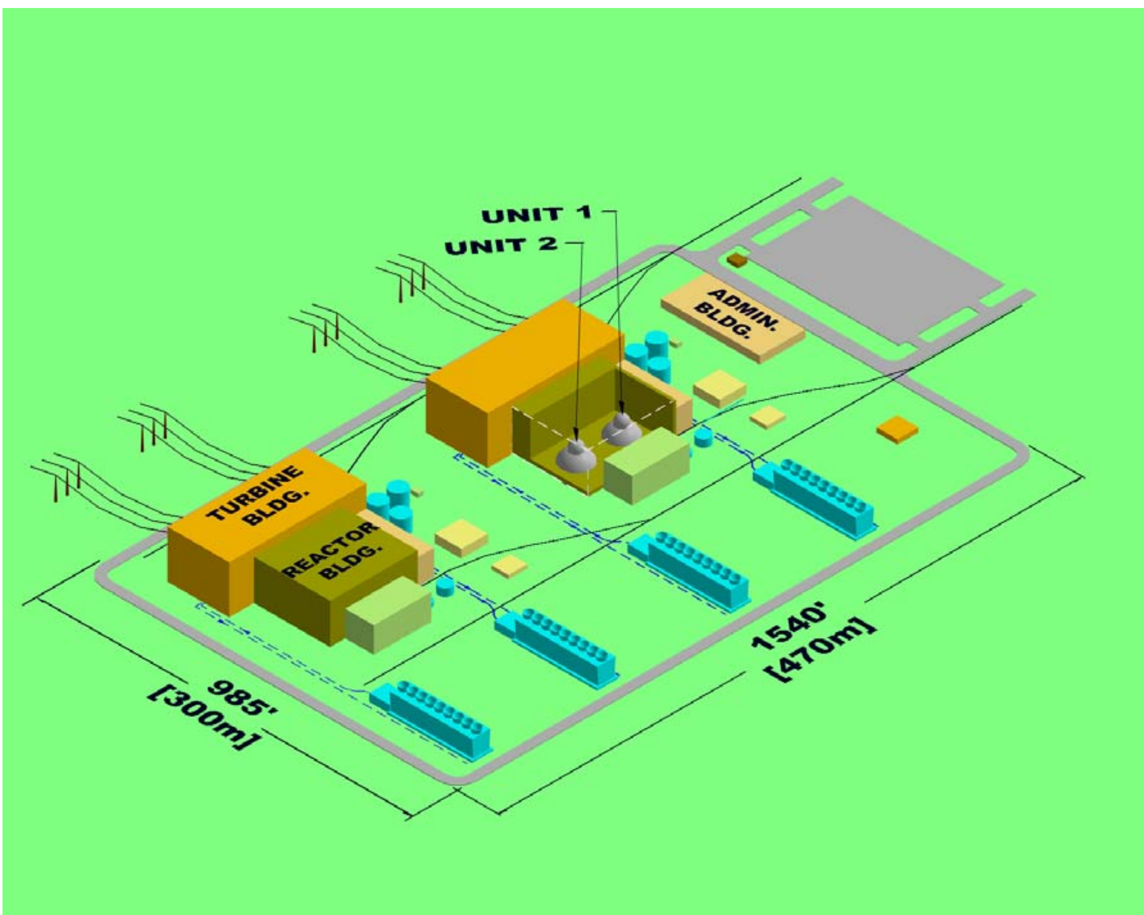


Figure 65 IRIS Two Twin-Unit Arrangement

increments, of the order of a few hundred MWe, may be however required by developing countries, which are projected to be an attractive market for IRIS, or even by utilities in developed countries which want to limit their financial exposure while allowing generation to be added as needed. In this case the independent single unit will be the answer.

In any event, the IRIS modularity, either twin unit or single module, allows construction of multiple units parks in a “slide along” manner, where generating capacity and cash flow is provided by the first unit, while subsequent ones are in construction.

Potable water availability is a looming major world-wide crisis and several studies indicate that in many developing countries the need for water is even greater than the need for electricity. Thus, a number of countries are considering nuclear plants for desalination (see, for example, periodic IAEA symposia on desalination). The now shut down BN-350 fast breeder reactor operated with associated desalination facilities for several decades on the Caspian Sea at Shevchenko (Aktau) in what is now Kazakhstan. Other nuclear desalination facilities operate in Japan, and India connected a desalination plant to two nuclear power plants in April ’02.

Studies have therefore been initiated to develop a co-generation IRIS version capable of dual electricity and water desalination. It is envisaged that only non-safety-related modifications will be required. OKBM has experience in designing nuclear plants for desalination, while CNEN and POLIMI have initiated economic and environmental evaluations.

Published cost estimates indicate an up-front cost to launch a small nuclear desalination plant between $50M and $60M, representing about 20% of the total cost. The costs of producing a cubic meter of fresh water from a 335 MWe nuclear power/desalination plant are estimated at 25 cents.

It has recently been reported[33] that in major developing countries such as China, India, Argentina, Iran, Turkey and Brazil, potential market demand has been assessed to be between 25 and 60 modules. IRIS is ideally placed to capture this market, particularly in developing countries, which may not have the necessary grid infrastructure, or which require multiple, diverse sources of power. Options being considered for IRIS are a true co-generation plant where low grade steam is used for desalination (distillation bottoming cycle), as well as a strictly electricity producing plant but where some of the electricity is used to run a reverse osmosis plant. In the second case, a rather obvious configuration is a twin unit where the electricity produced by one of the two modules is used for desalination.

Obviously any eventual solution is strictly application dependent. The first design to be performed will be for deployment in the arid Brazilian northeast. The capability for such direct input and feedback within the project is one of the unique advantages of the wide IRIS international consortium.

13. ECONOMICS

It is obvious that economy is an absolute imperative. If IRIS is not economically competitive with other energy sources, nuclear and non-nuclear, nothing else matters, regardless of how attractive are its technical and safety characteristics. Since IRIS is a small-to-medium size modular reactor, the classic economic of scale does not apply; actually, if IRIS were just a single module, it would make IRIS outright uneconomical. On the other hand, smaller modular plants have other advantages which will offset the penalties of scale. They are: increased factory fabrication; more replication; multiple units at a single site; improved availability; faster progression along learning curves; bulk ordering; better match to demand; smaller front end investment; reduced construction time; increased station lifetime; elimination of some engineered safety systems and simplification of others; design appropriate to the site. A discussion of each item can be found in Reference 4.

Another potentially large advantage of smaller units has surfaced during the mid-August massive blackout in the northeast of the USA, which required large nuclear plants to remain longer off the grids. Smaller plants provide less impact when off-line and they are easier and quicker to be returned to the grid following grid disturbances.

Work has been initiated to quantify the above economic advantages of smaller, modular plants. In a simplistic way, they will result in a multiplier less than one of the capital cost and cost of electricity calculated for a single module. While a bottom up module cost evaluation has to wait for completion of the preliminary design, a top down assessment was performed along with a sensitivity assessment of key factors in design and development.

A high level diagram of those factors influencing the value to a potential utility (see Figure 66) was developed to understand the decisions utilities face when choosing which type of design to invest in, e.g. a small modular reactor vs. a large conventional design.

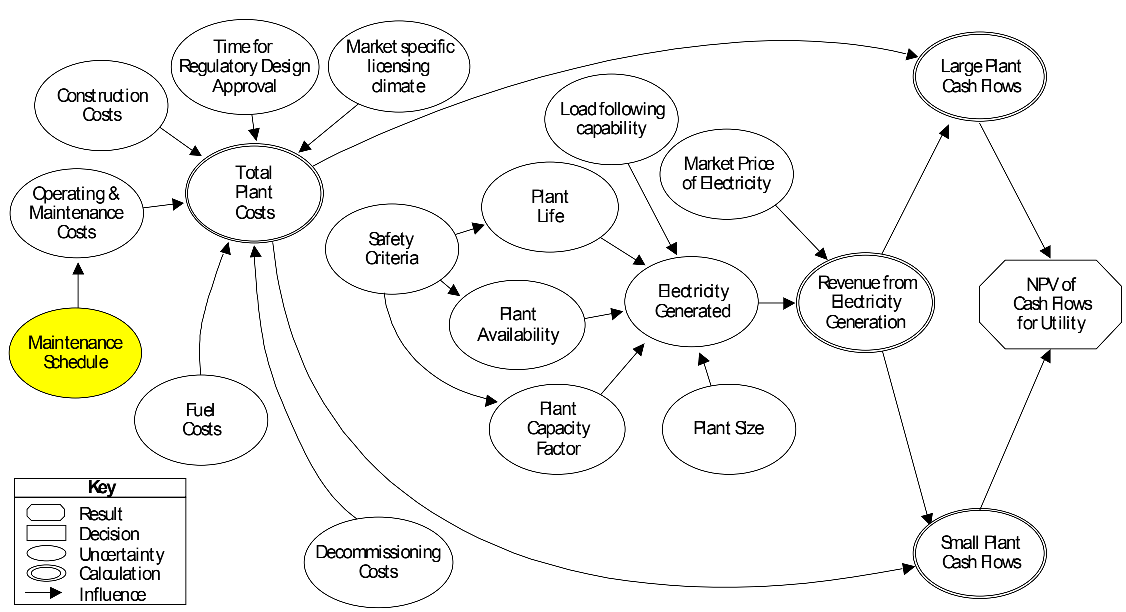


Figure 66 High Level Influence Diagram

Data were then collected for each uncertainty (ellipse in Figure 66), for each of the alternatives considered. The resultant pedigree of information forms the basis of the analysis and was peer reviewed by a group of senior executives, not directly involved with the IRIS project.

Extensive analysis has highlighted the importance of optimizing maintenance schedules and their impact on annual operating costs. Reducing these costs by the order of $1M/year is the equivalent of an additional $70M in lifetime value to the owning utility. Maintenance optimization over a four-year outage schedule has been discussed in Section 9. Advantages include increased capacity factor, reduced requirement for on site staff with further reductions in O&M costs, as well as the ability to build IRIS in remote locations with sparse local populations.

The basis of the assessment chosen for comparing alternatives is the lifetime Net Present Value (NPV) of cash flows. Each alternative was subjected to a rigorous analysis, to understand the implication and number of technical challenges and their ability to achieve the design and construction schedule, leading to market deployment early in the next decade.

The analysis focused on determining the optimal configuration for IRIS to establish generation costs ($/MWh) and Internal Rate of Return to the utility (IRR %) at alternative power ratings for IRIS. This was then combined with global market projections for electricity demand out to 2030, segmented into key geographical regions.

It resulted that the optimum single module size is about 335 MWe (in fact this study was a key factor leading to increase the IRIS power to 1000 MWt), with a construction period of three years or less and a minimum plant life of 60 years. As seen in Section 12, individual modules can be installed in a staggered fashion (three modules, equivalent to 1005 MWe) or built on site to match demand in pairs (two sets of twin units, equivalent to 1340 MWe).

The analysis context was to assess the viability of deploying an IRIS reactor (of varying electrical output) in 8 key geographic regions of the world:

- North America
- Western Europe
- Industrial Asia
- Eastern Europe / Former Soviet Union
- Developing Asia
- Middle East
- Africa
- Central and South America

Comprehensive financial modeling of reactor cash flows was used as the basis for comparing generation costs for different versions of IRIS and for conventional LWR designs. The analysis included a full sensitivity assessment of the key parameters (Figure 66), together with their supporting subset developed during financial modeling. A deterministic sensitivity analysis (Figures 67 and 68) ranked all parameters in their order of importance, focusing attention on those vital to success. The final area of modeling completed a probabilistic analysis of the top 10 parameters (as identified by the deterministic sensitivity), to understand how changes in these parameters would impact overall NPV and generation costs.

Output from the IRIS financial modeling indicates that market clearing price ($/MWh), construction costs ($M) and reactor power output (MWe) are the key factor in driving value. A commercially sized IRIS (335 MWe) is capable of competing in all world markets, with generation costs of approximately 30 $/MWh. The modular design and smaller output of IRIS is particularly suited to the developing markets. There is a major opportunity to install new nuclear generating capacity in the developing countries because their electricity consumption between 2000 and 2030 is predicted to grow at twice the rate of that in the developed nations.[34] The staggered installation approach also enables utilities to match their investment programs with rises in demand for electricity, minimizing their financial exposure. It also avoids disruption of local market conditions, which could occur, e.g., when connecting a single large plant of over 1000 MWe capacity.

Table 16 presents a summary of the base case lifetime net present values ($M) for a site consisting of 3 IRIS modules each rated at 335 MWe. Financing periods of 10, 20 and 30 years are compared to highlight the impact of alternative approaches.

Table 16 Summary of Base Case Values to the Utility ($M)

	Geographic Regions			
Title Cash flow NPV	NA, CSA	WE, EEF, ME, AF	DA	IA
30 Year Finance	3,030	4,030	5,020	6,020
20 Year Finance	3,330	4,330	5,320	6,320
10 Year Finance	3,620	4,680	5,610	6,610

NA = North America
WE = Western Europe
IA = Industrialised Asia
EEF = Eastern Europe / Former Soviet Union
DA = Developing Asia
ME = Middle East
AF = Africa
CSA = Central and South America

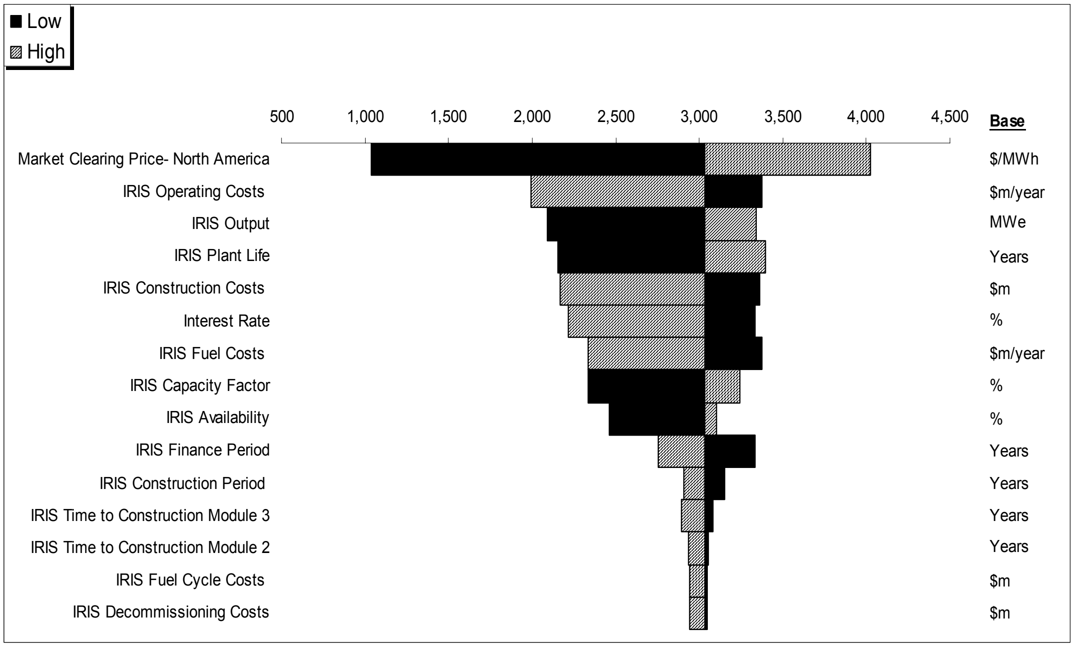


Figure 67 Key Input Parameters Lifetime Net Present Value ($M)

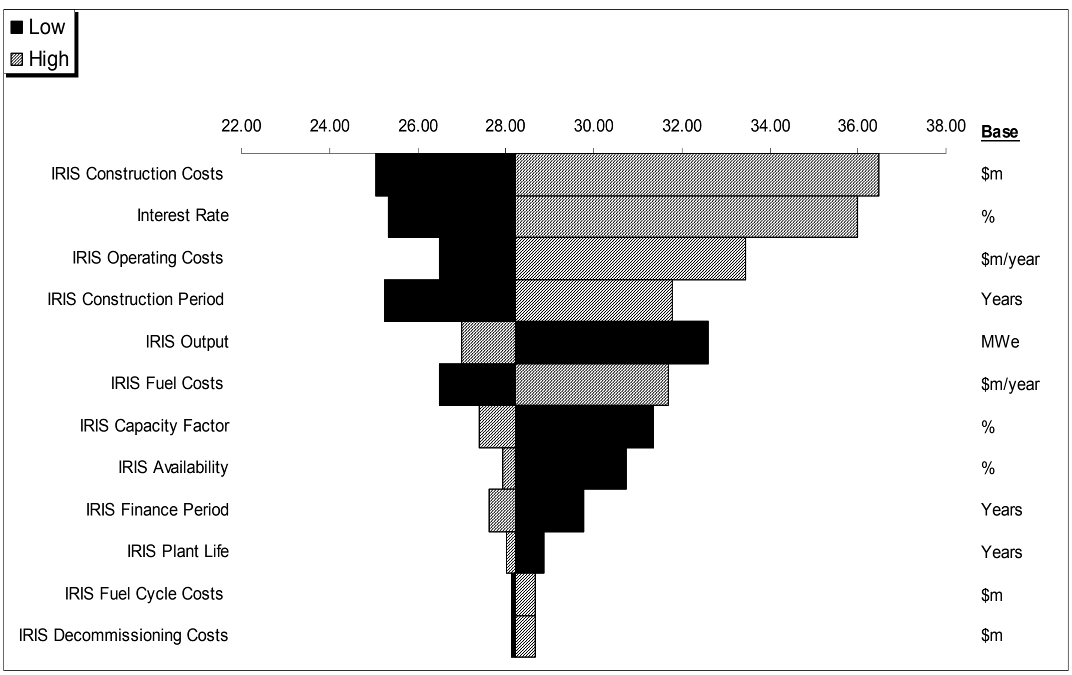


Figure 68 Key Input Parameters Generation Costs ($/MWh)

Figure 69 illustrates the probabilistic assessment of the likely range of generation costs, based on the top 10 input parameters as identified by the deterministic sensitivity. The expected cost, a single number that can represent the probability distribution shown in Figure 69, is 30.0 $/MWh. There is an 80% chance that generation costs will be in the range $22.0/MWh to $39.0/MWh, as shown by the 10% and 90% confidence limits.

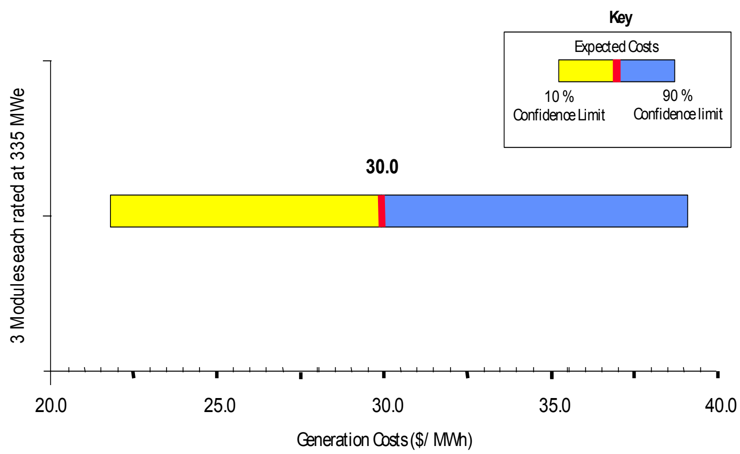


Figure 69 Probabilistic Analysis of Generation Costs

Based on the data set for an Nth-of-a-kind plant and including a full lifetime analysis of all costs and revenues, the major components of generation costs required to achieve a 20% Internal Rate of Return (IRR) are shown in Table 17. To further highlight the competitiveness of IRIS and its ability to compete over a broad range of market conditions, Figure 70 shows a comparison of generation costs with IRR over the range 10% to 30%, for 10, 20 and 30 year finance periods. All data are for a site in North America, having three IRIS modules each rated at 335 MWe.

Table 17 Generation Costs to Achieve an IRR of 20%

Cost Category	$/MWh
Construction (Financing Charges)	17.8
Operating & Maintenance	5.2
Fuel	3.4
Decommissioning	1.0
Fuel Cycle Costs	1.0
Total	28.5

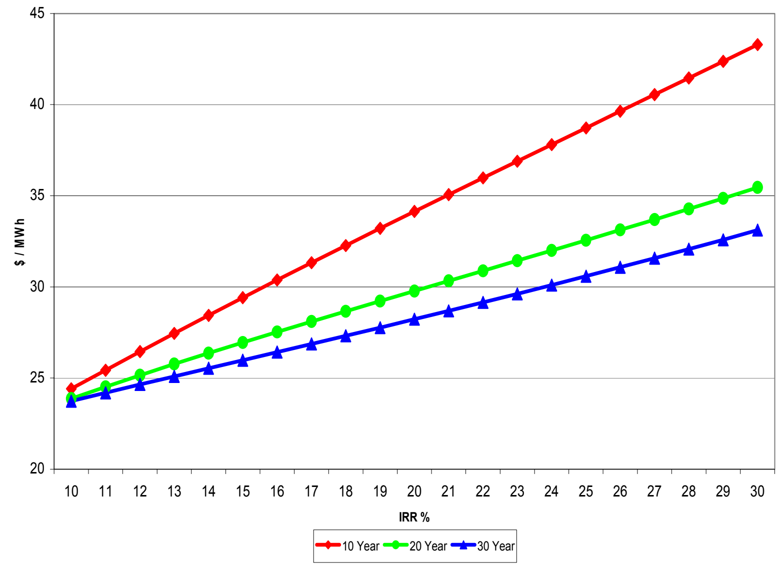


Figure 70 Generation Costs ($/MWh) versus Internal Rate of Return

This sensitivity analysis indicated that lifetime Net Present Values (from a utility perspective) are driven by uncertainties in:

- Market Clearing Price
- Finance Rate
- Major Plant Parameters
- Finance Period

These factors dominate the traditionally held industry view that all efforts should be focused on reducing Construction Costs**.**

The modular staggered design of IRIS allows utilities to match their build programs with capacity demands. It avoids issues of depressing local market clearing prices, which could occur when connecting a large plant grid. Financing charges can also be stretched and effectively managed, minimizing exposure to fluctuating economic conditions.

The current economic analysis demonstrates that IRIS is able to compete in all geographic regions with other nuclear designs and other energy forms of producing electricity. This competitive position will be further enhanced as the design develops and uncertainties are comprehensively quantified and then eliminated. Attention should in the first instance focus on improving the following major plant parameters:

- Capacity Factor
- Operating & Maintenance Costs

- Constructions Costs
- Fuel Costs
- Plant Life

Except for the construction costs, they have already been positively addressed in the IRIS design, as shown in previous sections.

Investing in an IRIS plant will provide a Utility with a commercially competitive IRR. Over the range 10% to 30% (Figure 70), generation costs for a 30-year finance period vary between 24.0$/MWh and 33.5 $/MWh. This is well within the range of market clearing prices forecasted to remain at or above 40 $/MWh.

14. PRE-APPLICATION LICENSING

As soon as IRIS focused on becoming a commercial project, design certification was immediately recognized as the qualifying issue. Thus, when IRIS was included in the ESP program, Westinghouse decided to initiate the licensing process by taking advantage of the recently instituted NRC pre-application licensing program. After various information meetings with the commissioners, staff and ACRS, the IRIS pre-application process was formally initiated in October 2002.

14.1 APPROACH

As previously mentioned, the rather ambitious target of the IRIS development and commercialization is deployment of the first unit in the 2012-2015 time frame. A critical step in attaining this goal is to successfully obtain design certification (DC) in 2008-2010, which would imply an immediate start of the process.

However, this is not possible for technical and programmatic reasons. First, the IRIS design, although quite advanced, has not reached yet the level of detail and the breadth of analyses required for design certification. Also, Westinghouse is currently pursuing DC for the advanced passive plant AP1000 and, rather obviously, Westinghouse does not want to stretch its and NRC resources by pursuing two DCs at the same time. It is expected that AP1000 certification will be successfully attained in 2005, which is therefore the earliest time when the IRIS DC can be started.

According to the, again ambitious, schedule this leaves three to five years to obtain DC for IRIS. This is not impossible, but it is certainly challenging. A sound strategy and planning is thus needed to successfully meet such challenge. The IRIS licensing approach is consequently articulated over the following premises:

1. IRIS will rely as much as possible on the successful attainment of DC by the Advanced Plants (AP) AP600 and AP1000. This means that wherever feasible and advantageous, IRIS will adopt the same design solutions as AP or current LWRs.

2. IRIS will immediately focus on the items where it differs from AP, such as the integral configuration and the safety-by-design.

3. IRIS will take advantage of the recently instituted pre-application licensing program to obtain NRC feedback on long term and novel items.

Thus, to complete design certification by 2008-2010, IRIS must first complete by 2005 its pre-application licensing, positively addressing the items identified in premise 3. IRIS will then be well positioned to move rapidly through DC by virtue of premise 1, coupled with having already addressed and well defined the issues of premise 2.

To meet the first premise/objective, the IRIS project in the second year (2001) of its existence underwent a major reassessment of its design to replace with current technology those very advanced features which were not intrinsically critical to attain the core IRIS objectives of enhanced safety and competitive economics. The results of this reassessment are summarized in Table 18.

Table 18 Evolution of IRIS Core Design

Characteristic	Original Concept	Current Reference
Core life	Up to 15 yrs	3-4 yrs
Lead rod burnup	$\geq$ 100,000 MWd/t	< 62,000 MWd/t
Fuel enrichment	< 20% fissile	Up to 4.95% enriched UO_2
Core configuration	Tight lattice, hexagonal	Standard square lattice
Neutron spectrum	Epithermal	Thermal, enhanced moderation
Control	No soluble boron	Limited soluble boron
Heat removal	Substantial natural circulation, with controlled boiling	Multiple (8) pumps, with subcooled conditions

First of all, the fuel enrichment and burnup is now within current licensing and fabrication limits. The IRIS fuel assembly is very similar to the standard Westinghouse 17x17 robust fuel assembly. Thus, IRIS fuel presents absolutely no licensing issues. Advanced core configurations to attain a harder neutron spectrum and thus stretch the core life were discarded; similarly discarded was the presence of boiling which had the purpose of enhancing natural circulation. The rationale was that both features required extensive development and the possible advantages did not offset the increased cost and delay.

Similarly, the no soluble boron core design, although quite attractive, was replaced by present technology, but with a more limited quantity of boron, when the IRIS module power output was uprated from 300 MWt (~100 MWe) to 1000 MWt (~335 MWe).

At the conclusion of this reassessment, the IRIS design was firmly rooted on proven LWR technology, but newly engineered to a safety-by-design integral configuration. Thus, the project next concentrated its focus on the unique characteristics of the integral design and how they are factored into the safety-by-design approach.

The safety-by-design, the IRIS passive safety systems and their implication have been presented in Section 10.1 and 10.2. The consequences of the safety-by-design on the IRIS outlook to licensing are two-fold. On one hand the significant enhancement of the defense in depth, coupled with passive safety system designs similar to AP will provide a strong background to attain DC once that the novel features of the IRIS integral design are adequately investigated both analytically and, where necessary, experimentally. At the same time and conversely, the added dimension to the defense in depth provided by the safety-by-design, when combined with an appropriate risk-informed approach would allow IRIS to attain more ambitious objectives such as demonstrating no need for off-site emergency response.

Consequently, the IRIS pre-application licensing is focused on two major areas (or phases, since they will be done sequentially):

A. NRC review and feedback on the IRIS experimental plan to investigate the characteristic aspects of the integral design and its safety effects and to confirm the predicted IRIS response to design transients and accident sequences.

B. NRC review and feedback on the IRIS approach to "enhanced licensing objectives" such as no requirement for off-site emergency response, through risk-informed regulation.

These two areas of the IRIS pre-licensing application are discussed in the following sections. It must be pointed out that at the time of this writing the project has completed the submittal to NRC of the necessary documentation to proceed with the first formal review meeting, expected to occur in December 2003.

14.2 PRE-APPLICATION ACTIVITIES

14.2.1 Review of IRIS Experimental Program

It goes without saying that a comprehensive and exhaustive test campaign is a necessary requisite to a successful DC. It is also evident that testing is also one of the most costly and time consuming items. Thus, the optimum is to successfully conduct only the necessary and sufficient set of tests to confirm the design and safety characteristics of IRIS.

The purpose of Phase A of IRIS pre-application licensing is therefore to present to NRC the test plan and obtain NRC feedback regarding its adequacy; the ultimate goal is to attain a mutually agreed test program before starting the test campaign, thus substantially saving time and money.

Tests to be conducted are those which address the unique aspects of the IRIS design. The first series of tests will be those confirming the design characteristics of the integral components such as the steam generators, pumps, pressurizer and control rod drive mechanisms. Both individual components and integrated effects (e.g., interaction between pumps and steam generators) will be conducted. The second series of tests will address integral effects, i.e., safety tests investigating IRIS response to design basis accidents, and confirming the analytical predictions. Confirmation of the safety-by-design will be obtained; a typical example of integral tests to be conducted is the investigation of the coupled behavior of the vessel and containment during a small-medium LOCA which intrinsically limits the break flow.

Finally, IRIS will take advantage, whenever applicable, of the results of previous tests from both the AP and SBWR programs (IRIS response to some accidents is similar to BWRs).

Thus, in this phase the IRIS project will seek NRC feedback and eventual agreement on 1) adequacy of proposed test program and 2) applicability of previously conducted and existing tests.

Documentation to allow NRC to conduct such review includes:

- IRIS Plant Description Document, which provides a detailed overview of the IRIS design. According to the spirit of this pre-application, the document was written focusing on the IRIS specific characteristics, addressing only cursorily or not at all what is very similar to AP. This document has been transmitted to NRC.

- IRIS Safety Analyses Document, providing a discussion of the IRIS response to typical safety sequences considered in Chapter 15 of the SAR. Documentation of the RELAP models adopted in these analyses is also included. This document, in two volumes, has been transmitted to NRC.

- PIRT (Phenomena Identification and Ranking Table) Document. This is the first step in the EMDAP (Evaluation Model Development and Assessment Process).[35] The IRIS PIRT shows that many high ranked phenomena for standard PWRs do not have to be considered for IRIS because they are eliminated by design. In addition, because IRIS is a PWR, with some BWR characteristics, no new phenomena unique to IRIS have been identified. The PIRT has also helped identify where data for unique IRIS design features are needed to reduce analysis uncertainty. Submittal to NRC is scheduled for March 2004.

- IRIS Scaling Analysis to provide a detailed overview of the various stages in specifying similitude conditions for the testing program. The IRIS Scaling Analysis Document, Part I has been provided to NRC and gives a detailed overview of the first two stages, Stage 1 - System Decomposition and Stage 2 - Scale Identification of the Hierarchical, Two-Tired Scaling Analysis. [36]

  A physically based decomposition (Stage 1) establishes a hierarchical architecture for the system. The system is subdivided into subsystem components and further into modules. Various constituents (materials) are inside the modules. Some constituents are in several phases (for example, water could be in liquid, or gas phase), and each phase can be in various geometrical configurations (for example, the liquid phase of water can be in the shape of droplets, liquid films, ponds, pools, or bulk coolant stream). The second stage provides hierarchy for the volumetric concentrations, area concentrations, residence times and process time scales. Two spatial and two temporal scales are associated with each transfer process to account for the effect on two constituents (or phases) separated by the same transfer area (but occupying different volumes and having different flow rates).

  Part II of the document is currently being prepared to address Stage 3 - Top-Down System Scaling Analysis and will be submitted about December 2003. It provides the adequate conservation equations so that scaling groups and characteristics time ratios can be calculated. Then the scaling hierarchy and identification of the important processes to be addressed in the last stage will be obtained.

- After NRC review and concurrence with the above documents, the IRIS test program will be prepared. It will include: tests to be conducted; test plan; proposed test facilities; and, test matrix. Its submittal is scheduled for Summer 2004.

- Finally, the last document will be Part III of the Scaling Analysis, i.e., Stage 4 – Bottom-Up Scaling. This will be prepared after resolution of NRC comments on the test program and selection of the appropriate test facilities.

14.2.2 Review of IRIS Approach to Risk-Informed Regulation

Because of the significantly added DID, thanks to its safety-by-design, IRIS is expected to meet with very ample margin the current licensing requirements.

This, coupled with its reliance on the precedents of AP600 and AP1000 successful DCs, will bode well for DC attainment by IRIS as well. However, IRIS intends to take advantage of its ample margin by relaxing the current licensing requirements to include no need for off-site emergency response planning. The project intends to use a highly risk-informed approach to attain this goal through use of PRA to guide final design and safety analyses.

Risk-informed can be represented as starting with the current design and licensing basis of a particular plant and justifying improvements to the regulations or maintenance of that plant based on risk assessments significantly driven by a PRA. The underlying approach is to justify changes from current regulations and licensing commitments, with defense-in-depth remaining as the primary basis for design analysis and safety reviews.

In a most recent document,[37] the Advisory Committee on Reactor Safeguards (ACRS) discusses two models on the scope and nature of defense-in-depth. In the structuralist model, “defense-in-depth is primary, with PRA available to measure how well it has been achieved”. In the rationalist model, “the purpose of defense-in-depth is to increase the degree of confidence in the results of the PRA or other analyses supporting the conclusion that adequate safety has been achieved.” The ACRS stated that “what distinguishes the rationalist model from the structural model is the degree to which it depends on establishing quantitative acceptance criteria, and then carrying formal analyses, including analysis of uncertainties, as far as the analytical methodology permits.”

Because of its reliance on the added DID dimension of safety-by-design to drive its more ambitious licensing objectives, the proposed approach for IRIS could be termed “highly” risk-informed to signify its enhancement over the current structuralist approach. This proposed approach is consistent with the “new regulatory framework” recently proposed by NEI.[38]

Table 19 indicates the major differences between the structuralist and rationalist approaches and the approach proposed for IRIS.

Table 19 Comparison of Various Risk-Informed Regulatory Processes

Structuralist	**IRIS**	**Rationalist**
Deterministic	**Enhanced DID with Focused PRA**	**Probabilistic**
Risk-Informed Approach	**Highly Risk-Informed Approach**	**Risk-Based Approach**
Start with current designs and regulatory approvals	Review current regulations and Establish high-level design and safety criteria	Develop new design and regulatory process
Justify risk-informed changes	Apply PRA to design and licensing to achieve “stretch” goals	Use probabilistic criteria to assure safety
Use defense-in-depth as primary means of assuring safety	Enhance defense-in-depth and resolve new issues with NRC staff	Use defense-in-depth and safety margins as needed

The IRIS licensing plan would use a highly risk-informed approach to address “stretch” issues such as:

- Reducing the Emergency Planning Zone (EPZ) to the plant exclusion area, thus fulfilling the DOE goal of no off-site emergency response
- Reducing control room staffing requirements for multi-unit sites
- Reclassification, elimination, or re-definition of deterministic design basis events
- Reducing Severe Accident Mitigation Alternative (SAMA) Systems and/or Analyses
- Extending the containment integrated leak rate test interval
- Extending maintenance outage intervals

The elimination of the emergency response requirements outside the exclusion area will benefit both developed and developing countries, as it provides:

- A persuasive argument for increased public acceptance
- An economic relief for utilities
- The possibility to site IRIS closer to population centers, thereby decreasing transmission and infrastructure costs

PRA analyses are currently in progress. Even though it is still early, preliminary results indicate that that the IRIS core damage frequency is orders of magnitude lower than current LWRs and substantially lower than AP1000. Thus, it is expected that the IRIS PRA will show that the safety-by-design approach reduces the probability of severe consequences to the extent that "emergency plans and procedures" can "be dependent upon risk information" and the need for emergency planning can be eliminated.

This second phase of the pre-application is scheduled to be completed by mid-2005 with the goal of reaching a consensus on the IRIS risk-informed approach and its implications for licensing.

15. CONCLUSIONS

The IRIS project design progress in its first four years through October 2003 has been presented here. IRIS is of course quite different from current LWRs because of its integral configuration, but it has also unique characteristics which differentiate it from other integral designs.

The most striking is the IRIS safety-by-design approach, which has been embodied into a design methodology that has guided the entire design process. The enhanced safety of IRIS has gone hand in hand with a simpler, more economical design. Moreover, because of the vastly enhanced defense in depth, IRIS' objective is to be licensed without the requirement for off-site emergency response planning. The favorable economic, public acceptance and market impact effects will be extremely significant.

The project firmly believes that the simpler, straightforward IRIS design, coupled with its modular configuration and limited financial outlay requirements, will make IRIS very competitive on the power market. Future work will focus on quantifying and substantiating this belief.

Another IRIS discriminant is its international approach; the IRIS design fits both developed and developing countries, as evidenced by the consortium membership.

NRC pre-application licensing is in progress, quite a remarkable accomplishment for a barely four year old project. Its successful conclusion will be the immediate objective for the next two years.

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Beatty, J.M, N. G. Arlia, "G***eneration IV Reactor Inspection and Monitoring Needs Assessment***", Proc. American Nuclear Society Robotics and Remote Systems Conference, Seattle, 4-8 March 2001, ANS (2001).

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APPENDIX B

IRIS GRADUATE THESES

IRIS GRADUATE THESES				
SCHOOL	**STUDENT**	**DEGREE**	**TITLE**	**DATE**
Polytechnic of Milan, Italy	Luca Oriani	PhD	Thermal Hydraulic Studies on the International Reactor Innovative and Secure (IRIS) and Its Original Containment Concept	11/2001
	Antonio Cammi	PhD	Risk Informed Approach in the Safety by Design Concept for IRIS	7/2003
	Fulvio Trudi	PhD	Mechanical Design of IRIS Special Components	7/2003
	Andrea Cioncolini	PhD	Dynamic Thermalhydraulic Analysis of IRIS	3/2005
	Luigi Corleto	MS	Innovative Containment for Integral PWR	12/1999
	Caterina De Masi	MS	Methodology and Results for the Technical and Economical Optimization of Integral Reactors	12/1999
	Paolo Viganò	MS	An Original Hydraulic System for Nuclear Reactors Control Rods	3/2000
	Andrea Cioncolini	MS	Modular Steam Generators for the IRIS Project: Concept Solutions and Preliminary Experimental Evaluation	10/2000
	Andrea Oblatore	MS	An Analytical Model of Hydraulic Control Rods and Design of an Experimental Testing Facility	12/2000
	Aurelien Tissot	MS	Innovative Solutions for IRIS Steam Generators	3/2001
	Fabio Berra	MS	An Innovative Containment type for Integral Reactors	4/2001
	Francesco Maldari	MS	An Innovative Shielding and Decommissioning Strategy for Integral Layout Reactors	4/2001
	Fabio Famiani	MS	Steam Generators and Balance of Plant Optimization for IRIS	12/2001
	Francesco Ganda	MS	Neutronic Design of a Once Through Fuel Cycle for IRIS	6/2002
	Marco Colombo Claudio Rizzo	MS MS	Experimental and Theoretical Analysis of the Internal Hydraulically Driven Control Rods Concept (joint thesis)	6/2002

IRIS GRADUATE THESES				
SCHOOL	**STUDENT**	**DEGREE**	**TITLE**	**DATE**
	Lorenzo Santini	MS	Thermalhydraulic Aspects of Helical Steam Generators for Nuclear Plants	12/2002
	Andrea Cremona	MS	Thermo-mechanical issues of Innovative IRIS Steam Generator	7/2003
	Elisa Urbinati	MS	Preliminary Dynamic Modeling of the IRIS Once-Through Steam Generator for Control Purposes	7/2003
	Enrico Magnani	MS	CFD study of the IRIS downcomer and internals configuration	10/2003
	Andrea Maioli	MS	A Risk-Informed Approach to the IRIS Safety	10/2003
	Alessia Vitulo	MS	Development of the Hydraulically Driven Control Rod system	12/2003
	Rossella Bongiorni	MS	Numerical Analysis and Preliminary Experimental Test on a Helical Coil Tube Steam Generator	12/2003
	Andrea Ghilardi	MS	Simulation and Control of the IRIS Integral Reactor	3/2004
	Diego Conti	Undergrad Thesis	Thermodynamic and Economic Evaluation of Integral Reactors for Electricity and Potable Water Production	7/2003
MIT, USA	J. Saccheri	PhD	An Optimized Epithermal Light Water Reactor Core Design	5/2003
	A. Romano	MS	Novel Fuel Geometries for Generation IV IRIS Reactor	2/2001
	M. Galvin	MS	Maintenance Cycle in Advanced Light Water Reactor Design	10/2001
	J. Saccheri	MS	Core Design Strategy for Long-Life Epithermal Water Cooled Reactor	5/2002
	S. Blair	MS	Thermal Hydraulic and Economic Analysis of IRIS	5/2003
	Zhongtao Wu	Undergrad Thesis	Neutronic Analysis of Tight Cores	5/2003
University California at Berkeley, USA	T. P. Lou	MS	Feasibility Study of Long-Life Cores for “IRIS” with Small Burnup Reactivity Swing	6/2000
	Hiroshi Matsumoto	MS	Exploration Study of Long-Life Cores for IRIS	6/2001

IRIS GRADUATE THESES				
SCHOOL	**STUDENT**	**DEGREE**	**TITLE**	**DATE**
University of Pisa, Italy	Fausto Franceschini	PhD	Study of Advanced Fuels for LWRs	10/2005
	Antonio Cipollaro	MS	Nodalization and Analysis of Thermal Hydraulic Transients of the Generation IV Nuclear Reactor IRIS by RELAP5/ Mod 3.2 Code.	10/2001
	Fausto Franceschini	MS	Neutronic Design of the IRIS Reactor Core	10/2002
	Alessandro Del Nevo	MS	IRIS Reactor Thermal-hydraulic Behaviour by RELAP5 Code	10/2002
	Gianni Ambrogi	MS	IRIS Reactor Thermal-Hydraulics	10/2003
Tokyo Institute of Technology, Japan	V. Barchevtsev	PhD	Synergistic Potential of PWR and PTGR in Achieving High Fuel Burnup	3/2001
	Y. Mizuno	PhD	Risk-Informed Design Approach to the IRIS Plant	3/2003
	T. Misawa	PhD	Direct Numerical Simulation of Detailed Temperature and Turbulent Velocity Distributions in a Subchannel of Tight Lattice Rod Bundles	3/2003
	E. Baglietto	PhD	Optimization of the Novel Fuel Geometry for a Future High Burn-Up Option Core of the IRIS Plant	9/2004
	S. Yamada	MS	Evaluation of DNBR of a Tight Lattice Core Option for the IRIS Plant by Subchannel Analysis Method	3/2003
	J. Kaneko	MS	Numerical Simulation of Large Bubble Behaviors between Subchannels of a Tight Lattice Bundle Option for the IRIS Plant under Transient and Accident Conditions	3/2004
	Y. Koike	MS	Risk-informed Design Approach to the IRIS Safety Systems	3/2004
University of Tennessee, USA	Wesley Williams	MS	Modular BOP Design for IRIS	5/2002
	Allan Wollaber	MS	Reactivity Control Concepts for IRIS	1/2003
	Vasanth Murthi	MS	Simulation of Factory Production of IRIS Modules	6/2003
	Martin Williamson	MS	Optimization of IRIS BOP Components	5/2004

IRIS GRADUATE THESES				
SCHOOL	**STUDENT**	**DEGREE**	**TITLE**	**DATE**
Ohio State University, USA	Andrew Kauffman	PhD	Design, Construction and Evaluation of In-Core Nuclear Power Sensors	2003
	Garret Wonders	MS	Feasibility Analysis and Preliminary Design of a Constant Heat Flux Power Sensor for In-Core Power Measurements	6/2001
	Keith Maupin	MS	Design, Construction and Evaluation of a Constant Heat Flux Power Sensor for In-Core Power Measurements	2002
	Dongxu Li	MS	An On-Line Degradation Monitor of a Constant Temperature Power Sensor	2002
	Daniel Mills	MS	A Research Reactor Test Facility that Simulates the IRIS Instrument Channel Thermal Environment	2002
University of Zagreb, Croatia	Tomislav Bajs	PhD	Coupled Code Analysis of the Primary System and Containment for IRIS Reactor	2004
	Srdjan Spalj	PhD	3D Neutronics and Thermal Hydraulic RIA Calculational Scheme for IRIS Core	2004
	Radomir Jacmenica	PhD	Optimization of Burnable Absorbers for IRIS Core	2004
	Kristijan Gergeta	MS	Loading Pattern Development for IRIS Core	2004
Polytechnic of Turin, Italy	Paolo Ferroni	MS	Analyses of Beyond Design Basis Accidents for IRIS	2005