Trade-Off Studies for Defining the Characteristics of the IRIS Reactor Core

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ABSTRACT

This paper presents neutronic and thermal-hydraulic trade-off studies performed to establish the core characteristics of the IRIS reactor, an integral configuration, light water cooled modular reactor of small-to-medium power (100-300 MWe/module) which is being developed by an international consortium led by Westinghouse. Briefly described are the results of analyses which led to the selection of enriched UO_2 and MOX fuel, with a fissile content of ~ 10% in an open lattice configuration yielding an eight-year straight burn core life, without shuffling or refueling. Full natural circulation was found to be impractical and not economical, but the IRIS reactor can be designed with a substantial degree of natural circulation thus providing passive safety in the case of loss of flow accidents.

INTRODUCTION

IRIS (International Reactor Innovative and Secure) is an integral primary system configuration, light water cooled modular reactor of small-to-medium power (100-300 MWe/module) which is being developed by an international consortium.

In 1999, Westinghouse, together with two American universities (Massachusetts Institute of Technology and University of California at Berkeley) and one Italian university (Polytechnic of Milano), submitted a winning proposal under the DOE Nuclear Energy Research Institute (NERI) program to develop a Generation IV reactor, i.e., a reactor which offers proliferation resistance, improved economics, enhanced safety and waste reduction. The proposed concept proved to be quite attractive as many organizations from around the world joined, under their own funding, in its development at various times from late 1999 to early 2001. Currently the IRIS consortium includes 13 organizations from six countries. The four original partners were joined in successive order by Japan Atomic Power Company (JAPC); Mitsubishi Heavy Industries (MHI), Japan; British Nuclear Fuel (BNFL); Tokyo Institute of Technology, Japan; Bechtel, USA; University of Pisa, Italy; Ansaldo, Italy; NUCLEP, Brazil; and, National Institute for Nuclear Studies, Mexico. The French CEA was also an active participant until October 2000 when it withdrew as a consortium formal member.

Prior papers have described the overall characteristics of IRIS as they developed [1-3] or have discussed specific technical areas and presented novel solutions [4-7]. Intent of this paper is to complement the above by addressing "how" the design team has arrived to the selected choices through a trade-off of various alternatives.

In conducting the trade-off studies the following requirements 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 was "safety by design" [7] where by design most accidents either cannot occur or will not have serious consequences.
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 drastically in the last year 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. In fact, IRIS is one of the plants currently being considered by NRC for evaluation towards design certification.

Thus the additional requirement was to adopt technical solutions which could be confidently deployed by 2010. More advanced solutions which require longer technology demonstration will still be pursued for eventual implementation in subsequent plants.

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

Before starting the discussion of the tradeoff studies, however, a brief description of the IRIS design will be given to provide the reader with the proper context. A detailed presentation of the IRIS design can be found in Ref. 2 and 3.

IRIS REACTOR DESIGN

IRIS features an integral vessel which houses the reactor core and support structures, core barrel, upper internals, control rod guides and drivelines, steam generators, pressurizer, heaters, internal spray located in upper head and canned motor reactor coolant pumps (see Fig. 1). 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 Fig. 1 is for a 300 MWt design. Hot coolant rising from the reactor core to the top of the vessel is being 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 vessel penetrations near the top of the core. The top location shown in Fig. 1 is the currently preferred position, but studies are still in progress to finalize this choice.

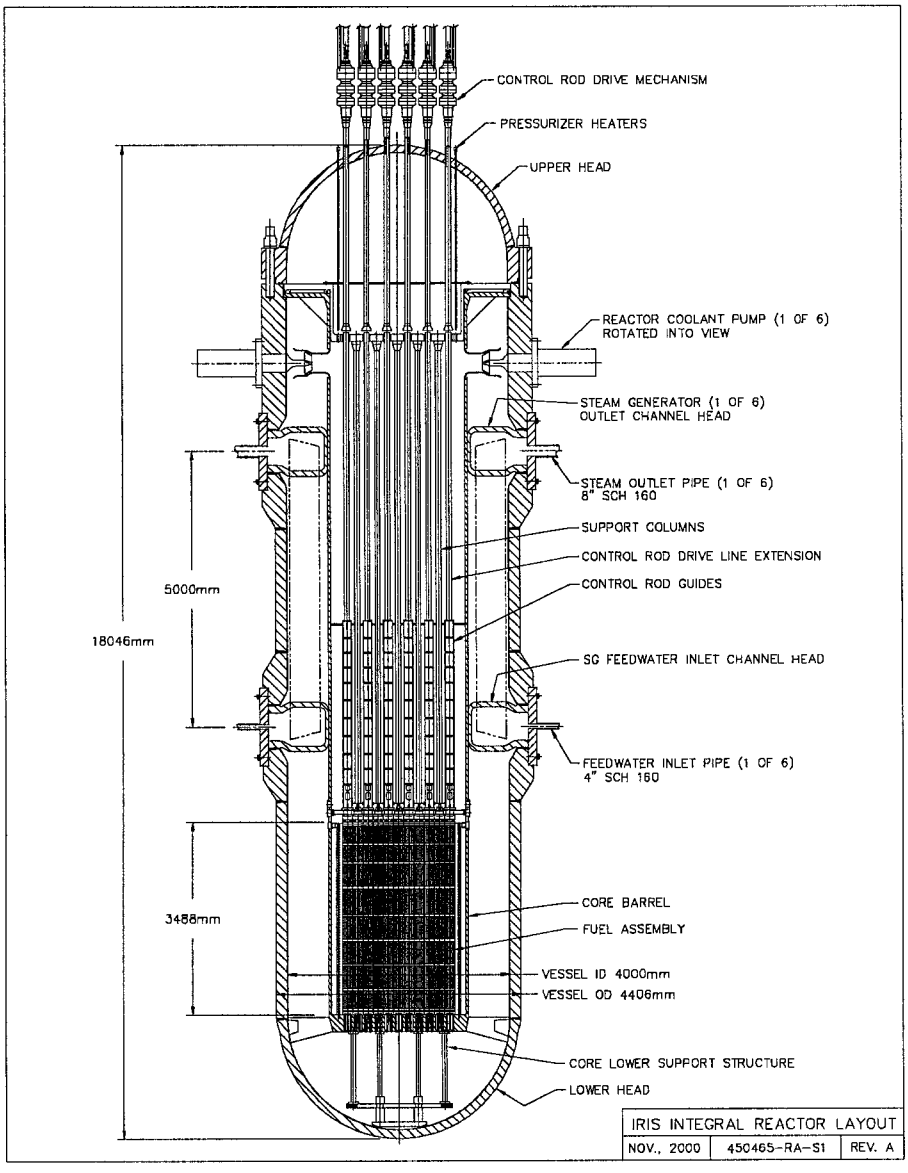

Fig. 1 IRIS Vessel Layout

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.

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 the next section, have acceptable, even though different, neutronic 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 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.

We envision IRIS not to be a static design, but to evolve with advances in technology and thus 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.

Fuel Lattice Selection

For light water cooled reactors, the fuel lattice is commonly represented by the fuel-to-moderator ratio, expressed either as the ratio of heavy metal (U+Pu) to hydrogen atoms, or, as the p/d ratio, where p represents the lattice pitch and d represents the cladding outer diameter. For unambiguous specification, assumed water density as well as the lattice geometry (square or triangular) need to be specified.

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 where the long core life is one of design objectives. Figure 2 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 Fig. 1, k-eff for the open lattice remains higher in the k-eff>1.0 region.

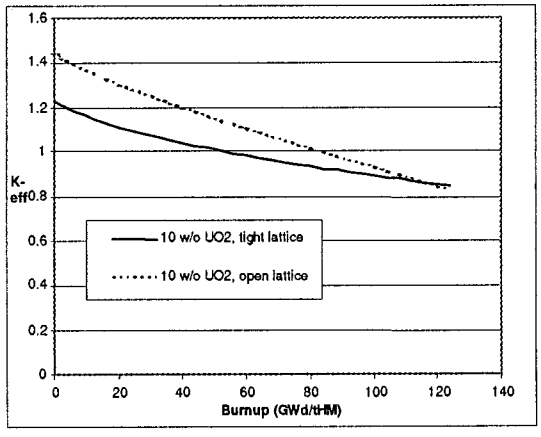

Fig. 2 Core Reactivity for Tight and Open Lattice (10 w/o ^{235}U in UO_2)

Optimum selection depends on other parameters, primarily fissile enrichment and fuel form (UO_2 or MOX fuel). We have performed a series of lattice calculations, calculating the infinite multiplication factor k-inf corrected for the neutron leakage, and translating it into achievable discharge burnup. Results for two different enrichments are reported in Figs. 3a and 3b, 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 higher discharge burnup in both cases (UO_2 and MOX). However, it is interesting to note that for higher fissile content, a tighter lattice provides higher discharge burnup for MOX fuel.

While 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 one 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.

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 capacity factor into the high nineties percentage. With long-life cores the maintenance outage becomes the limiting downtime interval which determines the capacity factor. IRIS will strive to reduce maintenance intervals through on-line diagnostic and maintenance as well as simple, reliable design of components. The current objective is to design IRIS to four-year scheduled maintenance intervals.

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 densities 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.

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 current 8-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 that 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.

Effect of lattice parameter p/d on the fast fluence and DPA (displacements per atom) is illustrated in Table I. Monte Carlo simulations were performed for lattices with different p/d ratios employing MOX fuel with 20% fissile content. 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.

Table I Fast (E>1MeV) neutron fluence and DPA (E>1MeV) in clad for 15-year core life (MOX with 20 w/o Pu, 3 kW/ft, based on BOL spectra)

p/d	Fast fluence Φ_{1MeV}	DPA
1.00	$6.0x10^{22}n/cm^2$	62
1.10	$4.7x10^{22}n/cm^2$	49
1.17	$4.0x10^{22}n/cm^2$	43
1.25	$3.4x10^{22}n/cm^2$	37
1.40	$2.6x10^{22}n/cm^2$	29
1.55	$2.1x10^{22}n/cm^2$	23

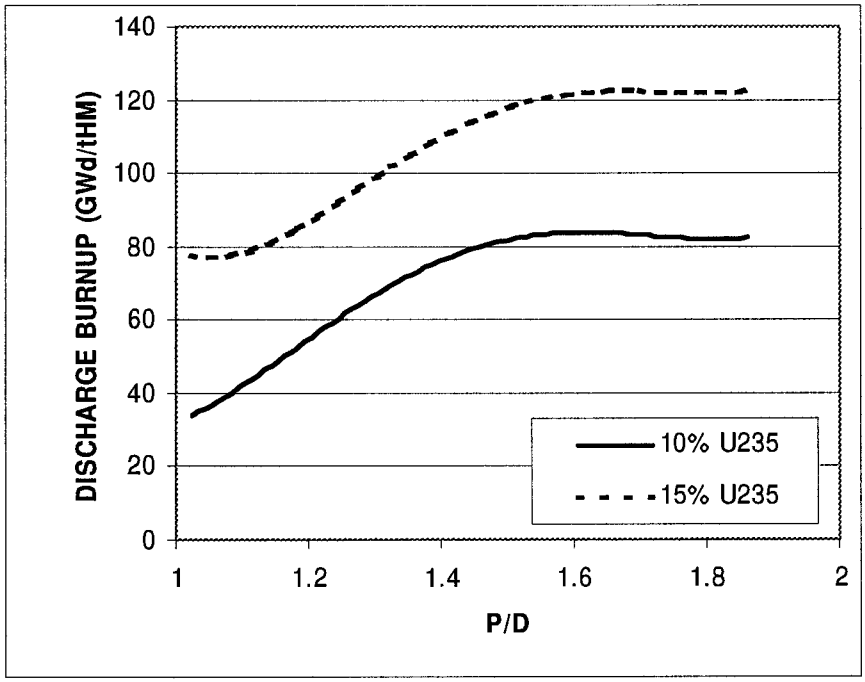

Fig. 3a Discharge Burnup as a Function of p/d for UO_2 Fuel

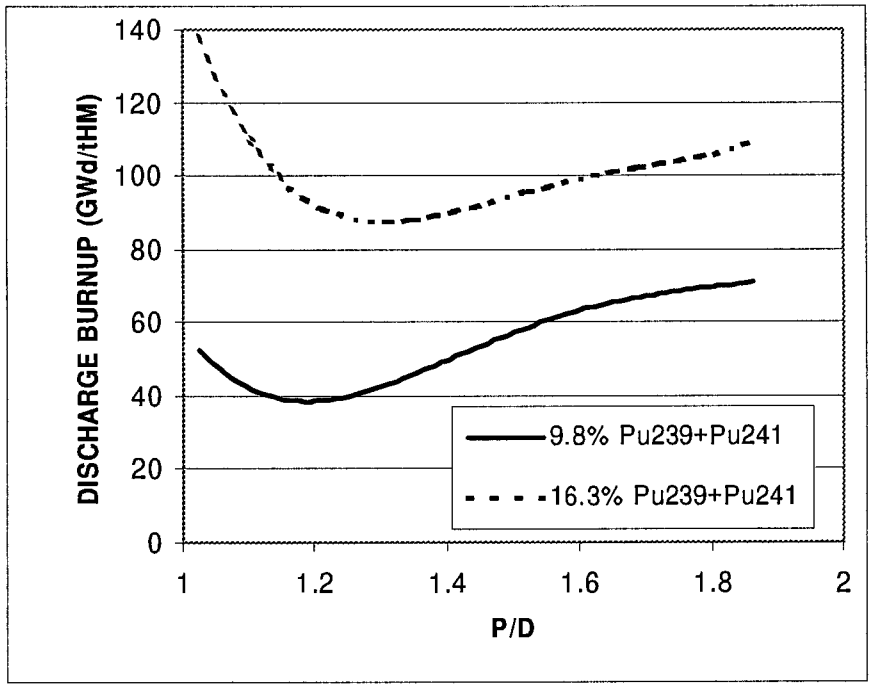

Fig. 3b Discharge Burnup as a Function of p/d for MOX Fuel

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 [8] and CAREM [9] feature full natural circulation to provide passive safety in the event of 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. More details on the code are reported in Ref. 5. 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 II.

Table II System Configurations Yielding Full Natural Circulation

		p/d = 1.45	p/d = 1.10	p/d = 1.05
Reactor power	MWt	300		
Secondary side pressure	MPa	6.0		
ΔT core	°C	55		
Average linear power	Kw/m	10.56	10.05	10.53
Vessel diameter	m	4.0	3.8	3.7
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, full natural circulation is highly impractical because of the required vessel height. However, even for the moderated open lattice core, the vessel height results to 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 II) 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 considered 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 lower, 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 is unfeasible. However, the advantages of natural circulation can be exploited in a design with only partial natural circulation aided by 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 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. It is our intent to design IRIS such that the amount of natural circulation flow ensures that design limits are not exceeded during loss of flow accidents.

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 pressure and efficiency as can be seen in Fig. 4. The core outlet temperature is practically dictated by the system pressure and core outlet quality, so is not too dissimilar from current PWRs. The selected inlet and outlet temperatures for IRIS were 292°C and 330°C, respectively. The value of the inlet temperature will be reassessed if the currently in progress transient analyses indicate that a higher degree of natural circulation is needed for safety reasons.

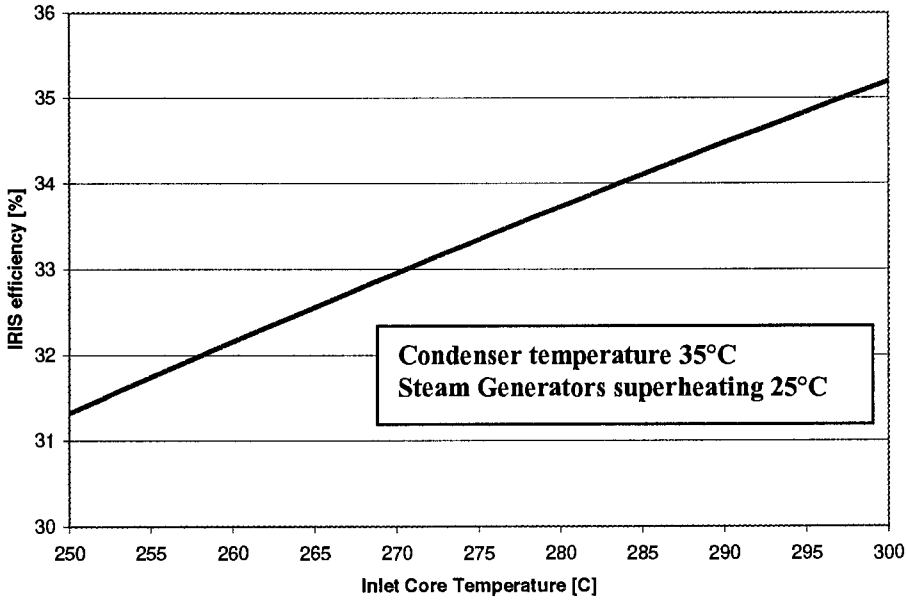

Fig. 4 IRIS Efficiency as a Function of Inlet Temperature

CONCLUSIONS

The IRIS design approach is to provide enhanced performance through innovative design solutions, but without the need for large technology development programs. The trade-off studies reported here helped to define the boundaries of a core design which, consistent with this approach, offers advancements in proliferation resistance through long core life without shuffling or refueling, economics through increased capacity factor, and safety through elimination of large LOCAs due to the integral primary system configuration and reduction of LOFA consequences due to high degree of natural circulation.

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