# PROGRESS IN DEVELOPING THE CONCEPT FOR THE IRRADIATION RESEARCH FACILITY

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## ABSTRACT

*At the 16th annual Canadian Nuclear Society conference, AECL presented the case for replacing the NRU reactor with an Irradiation Research Facility (IRF) to test CANDU® fuels and materials and to perform advanced materials research using neutrons. AECL developed a cost estimate of $500 million for the reference IRF concept, and estimated that it would require 87 months to complete. AECL has initiated a pre-project program to develop the IRF concept to minimize uncertainties related to feasibility and licensability, and to examine options for reducing the overall project cost before project implementation begins.*

## 1. INTRODUCTION

At the 16th annual Canadian Nuclear Society conference, Atomic Energy of Canada Limited (AECL) presented the case for replacing its ageing NRU (National Research Universal) reactor with an Irradiation Research Facility (IRF) [1]. The reference IRF concept would meet the Canadian nuclear industry's needs with various CANDU**-specific experimental facilities to test fuels and materials, and would provide facilities for advanced materials research using beams of neutrons. AECL has estimated the cost of the reference IRF concept to be $500 million, and estimated the schedule to complete construction to be 87 months. AECL is currently undertaking a pre-project program to further develop the IRF concept to minimize uncertainties related to feasibility and licensability, and to examine options for reducing the overall project cost before project implementation begins.

### 1.1 The Case for the IRF

The case for replacing NRU with a national dual-purpose IRF is based on the economic and technical benefits that Canada would continue to receive from a thriving nuclear industry and a dynamic advanced materials research community. For example, the total value of electricity and other goods and services produced by the Canadian nuclear industry is $6 billion annually (1993 estimate). In the recent program review, the Canadian government reaffirmed its confidence in the nuclear industry. Canadian scientists have also made significant contributions to the development of advanced materials by using neutrons to study the dynamics of matter and to confirm many theoretical predictions in condensed matter science. Those benefits will continue only if there is continued access to:

- Irradiation facilities to test CANDU advanced fuels and materials: The development of advanced concepts for the CANDU reactor (e.g., more passive safety systems, improved operation and maintenance, increased reliability, increased load factors, extended plant lifetime, and advanced fuel cycles) requires suitable experimental facilities to test new reactor fuels and materials under representative reactor conditions.

- Neutron beam research facilities: A source of neutrons is also essential to materials scientists who use neutron scattering techniques. Neutrons provided by NRX and NRU have facilitated world-class materials research (e.g., the awarding of the 1994 Nobel Prize in physics to B.N. Brockhouse for his work on determining the

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** CANDU (Canada Deuterium Uranium) is a registered trademark of AECL.

excitation properties in materials, and developing inelastic scattering techniques and instrumentation (i.e., triple-axis spectrometer)).

The detailed experimental requirements to support future development of the CANDU reactor (e.g., more passive safety systems, improved operation and maintenance, increased reliability, increased load factors, extended plant lifetime, and advanced fuel cycles) were defined by consulting with CANDU designers, researchers and utility representatives, and are described in various publications [1, 3].

The neutron beam facility requirements are described in publications [4, 5] by members of the Canadian Institute of Neutron Scattering and in the report [2] by the NSERC** -sponsored Committee on Materials Research Facilities.

## 2. DESCRIPTION OF IRF

The reference IRF complex [1] consists of the reactor in a CANDU-type containment building, a reactor-mock-up building, a guide hall with adjoining offices, laboratories and support facilities, an administration building and utilities and operations buildings. The estimated cost of $500 million for the reference IRF complex is based on the assumption that this is a complete facility that is located at an existing nuclear site.

The reference IRF concept [1] is based on a MAPLE*** -type reactor assembly in an $H_2O$-filled pool. The nominal reactor power is 40 MW, although the design power, and hence the flux level, will be optimized and fixed during the detailed design phase of the IRF project. There are two adjacent core segments with 18 sites each (Figure 1). Each core segment has twelve 36-element fuel bundles, four 18-element fuel bundles and two sites for materials irradiation rigs. The fuel element contains a core of $U_3Si_2$ in aluminum with an enrichment of 19.75 wt% $^{235}U$, and is enclosed in a co-extruded aluminum sheath. Annular hafnium absorbers surround the 18-element fuel bundles for reactivity control and shutdown. The core segments are surrounded radially by a reflector vessel filled with $D_2O$. A second diverse and independent shutdown system is provided by rapidly dumping the $D_2O$. The experimental facilities included in the reference IRF concept are:

- Horizontal fuel test facilities: three test sections with two or three CANDU bundles per test section, and one loop system per test section;

- Vertical fuel test facilities: two test sections for multi-element partial bundles with one loop system per test section;

- Blowdown test facility (BTF) loop: one BTF loop system to connect to the bottom horizontal test section;

- Materials irradiation facilities: four in-core sites with three or four inserts each, and four fast-neutron (FN) sites with four inserts per site or one corrosion loop per site;

- Hot cells: one three-compartment cell on the main level of the Reactor Building, one cell in the Operations and Utilities Building and one shielded facility for handling operations for the horizontal test sections;

- Service irradiation facilities: 10 vertical tubes including two for hydraulic rabbit systems and one for a pneumatic rabbit system; and

- Neutron beam facilities: 10 beam tubes, two of these for cold neutron sources, a liquid hydrogen ($LH_2$) cold neutron source, five cold-neutron guides, two thermal-neutron guides and six new spectrometers.

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*** NSERC (Natural Sciences and Engineering Research Council)
**** MAPLE (Multipurpose Applied Physics Lattice Experiment)

The performance of the experimental facilities has been estimated with the physics computer codes WIMS-AECL/3DDT [6, 7] and MCNP [8] to be:

- Peak unperturbed thermal-neutron flux: The peak unperturbed thermal-neutron flux in the $D_2O$ reflector has been estimated to exceed 4 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$.

- Horizontal fuel test facilities: With two natural uranium CANDU bundles per test section and $D_2O$ coolant, each bundle would produce ~630 kW in the bottom test section, ~950 kW in the middle test section, and ~680 kW in the top test section.

- Vertical fuel test facilities: The average linear element ratings from a seven-element partial bundle and $H_2O$ coolant is estimated to be ~37 kW/m with natural uranium.

- Materials irradiation facilities: In the core, the fast-neutron (E > 1 MeV) flux in representative QUATTRO rigs would exceed 1.3 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$ for a 150 mm length of zirconium alloy, and 1.0 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$ for a 450 mm length of zirconium alloy. Similar QUATTRO rigs in the FN sites would have fast-neutron fluxes that exceed 0.42 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$ for a 94 mm length of zirconium alloy, and 0.33 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$ for a 460 mm length of zirconium alloy.

- Beam tubes: The perturbed thermal-neutron fluxes at the entrances to the beam tubes are estimated to be about 2.5 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$ for BT1-4 and 1.8 x $10^{18}$ $n{\cdot}m^{-2}{\cdot}s^{-1}$ for BT5-10. The fluxes at the cold neutron source in BT9/10 were not calculated.

## 3. IRF PRE-PROJECT ACTIVITIES

Following completion of the reference IRF concept, AECL initiated a pre-project program to further develop the IRF concept to the point that any uncertainties related to feasibility and licensability are minimized before project implementation begins. By identifying and addressing these uncertainties, the risks of escalating costs and schedule extension after project commitment will be reduced. The pre-project program will improve the definition of the project scope so that appropriate resources can be applied and confidence is developed in the project cost estimates and the schedule to completion. The deliverables will include confirmation of technical feasibility and licensability, completion of a low risk project scope definition by reviewing the reference IRF concept to identify opportunities for cost and schedule reductions, and development of a plan for implementing scope changes.

### 3.1 Analysis and Testing Activities

The physics and thermalhydraulics codes used to model the IRF are not new but application to a new geometry with a high degree of heterogeneity increases the uncertainties in the ability of the codes to realistically estimate the behaviour of the reactor. Since completion of the safety analyses for use in a Preliminary Safety Analysis Report will require application of the physics and thermalhydraulic computer codes to predict the performance of the reactor under normal and upset conditions, the pre-project program includes activities to increase confidence in the physics and thermalhydraulics codes and modelling methods.

The physics activities include completing a review of the physics codes, building the WIMS-AECL/3DDT and MCNP reactor models, and performing calculations to provide input to the thermalhydraulics, safety analysis and design activities.

A review of the calculation methods for analyzing the IRF concept was performed. To conduct the review AECL invited participation from Ecole Polytechnique, Oak Ridge National Laboratory (ORNL) and Idaho Engineering Laboratory (INEL). The participants from Ecole Polytechnique have experience with WIMS-AECL and MCNP, and are knowledgeable about CANDU physics methods. The participants from ORNL and INEL have experience with the calculation methods (e.g., MCNP) applied to the conceptual design of the Advanced Neutron Source. The ORNL participant also has experience with modelling HFIR and the INEL participant has modelled ATR. The reviewers determined that the calculation methods are suitable for analyzing the IRF. They provided recommendations for improvements to the computer codes and methods. Some of the recommendations have been included in the pre-project program. Other recommendations require further study before an implementation plan can be prepared.

Detailed physics models are required to perform analyses to guide other pre-project activities. For example, the development of fabrication techniques for the IRF fuel with burnable poison is being guided by a physics analysis. This analysis is examining the concentrations of gadolinium or cadmium that would be required in the fuel core, between the fuel core and cladding or in the cladding to satisfy two criteria:

- limit available excess reactivity: The intent is to limit the range of movement of the control rods to between ~60% and fully withdrawn from the beginning to the end of a fuel cycle. A typical fuel cycle will involve four to five weeks operation at full power. Each refueling will involve replacing four 36-element bundles, two 18-element bundles and one FN bundle.

- reduce peak linear element ratings: By limiting the control rod movement, the axial flux shape will be flatter, and thus, will help to reduce the peak linear element ratings.

Physics calculations have also been performed to assess the impact from testing experimental CANDU fuel bundles containing slightly enriched uranium. The power output from each test section would approximately double for advanced CANDU fuel bundle designs if they are enriched to 2 wt% $^{235}U$. To avoid possibly de-rating the IRF during the irradiation of such enriched bundles, the feasibility of controlling test section power by introducing a neutron poison around each test section has been investigated. The assessment identified two potential methods for conceptual design studies, using soluble $^{10}B$ in a $D_2O$ annulus between the calandria tubes and the guard tubes or replacing the $CO_2$ gas in the annulus between the pressure tube and calandria tube with $^3He$ gas.

The thermalhydraulics activities include, building the CATHENA [9] models for the reactor primary cooling system (PCS) and the fuel test loop systems, performing calculations to support design and safety analysis activities, and completing the validation of CATHENA for research reactor conditions. Preliminary analyses have established the PCS performance requirements (e.g., flow, core $\Delta T$ and cooling during loss of flow transients). An experimental program is in progress to obtain improved critical heat flux data on single elements to extend the heat transfer database. Critical heat flux experiments on partial bundles are also in progress to provide data for validating CATHENA for research reactor conditions.

Techniques for fabricating the IRF fuel with a burnable neutron poison are being developed. Trial extrusions of the $U_3Si_2$-Al fuel core with gadolinium and cadmium, and of the cladding with gadolinium mixed in with aluminum have been performed. Samples of the extruded cladding have undergone corrosion testing. A method for verifying the homogeneity of the poisoned fuel using a neutron beam has been demonstrated.

### 3.2 Up Front Licensing Activities

The up front licensing activities are directed at reducing uncertainties in the safety and licensing process by obtaining agreement with the Atomic Energy Control Board**** (AECB) on safety and licensing requirements. The activities in progress include:

- preparing the Licensing Basis Document which outlines the safety philosophy, lists the top-level licensing and safety requirements for the IRF project and the design and operating criteria for ensuring that the IRF can be operated with acceptably low risk to the operators, to the general public, and to the environment;

- preparing the Safety Analysis Program which outlines the safety analysis work needed to support licensing activities (e.g., description of analysis methods and their validation and verification, a list of postulated

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**** The AECB will be renamed as the Canadian Nuclear Safety Commission when the Nuclear Safety and Control Act is enacted.

initiating events that could lead to an accident with potential radiological consequences, and an acceptance criteria);

- preparing Safety Design Guides for safety-related systems, shutdown systems, external hazards (e.g., seismic requirements and tornado and external missiles protection), and internal hazards (e.g., fire protection, radiation protection and pipe rupture protection) to describe the safety objectives and general design requirements to be followed in the design of the IRF (e.g., accounting for normal operating conditions, anticipated operational occurrences and design basis accidents);

- interacting with the AECB and other government agencies to define the requirements for an environmental assessment and to prepare the technical documentation that will be part of the environmental impact study; and

- preparing software validation documents for the physics and thermalhydraulics codes, including a Technical Basis Document that describes the physical phenomena to be analyzed for the safety analyses, a Validation Matrix document that describes the data sets that are available for demonstrating that the codes accurately represent a given physical phenomenon, and a Validation Plan that describes the process for validating the codes and the acceptance criteria for judging the agreement between the code predictions and the data sets.

## 3.3 Requirements for Containment

The AECB does not have specific requirements for severe accident analysis (events with probabilities $< 10^{-6}$ per year) or containment capability [10]. The AECB emphasizes severe accident management rather than designing systems to deal with severe accidents. To develop containment design requirements for the IRF, reviews were carried out on the containment design requirements for CANDU reactors and on the designs of containment and confinement systems for existing research reactors. Studies of severe accident phenomena are also in progress to:

- define methods for accommodating severe accidents,

- perform scoping analyses for representative reactivity accidents,

- develop methods for steam explosion analysis, to develop methods for hydrogen mixing and combustion analyses, and

- perform scoping assessments for beyond design basis accidents, and to identify additional work.

## 3.4 Design Activities

The design activities include developing the concepts for the reactor systems and experimental facilities to confirm their technical feasibility and to support the thermalhydraulics, physics and safety analysis activities, defining requirements for design verification studies, confirming the ability to manufacture unique components (e.g., the reactor vessel), and evaluating options for reducing the overall project costs and schedule.

The reference concept for the reactor structure (i.e., reactor vessel, inlet plenum and chimney) required alignment of large assemblies and concerns were raised about the feasibility of construction. A design study was carried out to identify options to reduce the complexity and cost. Manufacturers with experience in building complex assemblies were consulted to address feasibility issues. The design study has identified the following options for consideration:

- <u>eliminating the inlet plenum as a separate structure</u>: The reference inlet plenum is a short round tank with a tall central riser pipe which includes the grid plate for holding the fuel assemblies. The inlet plenum supports the reactor vessel. This arrangement raised concerns about alignment of the overall reactor structure. To address the concerns, a reactor vessel concept that includes two inlet pipes and a vertical pipe to deliver coolant to the core segments has been proposed. The grid plate would be attached to underside of the reactor vessel.

- simplifying the reactor vessel: The reference reactor vessel is a tall structure with an annular dump tank that fits around and rests on the inlet plenum. A review of the reference concept for the reactor vessel identified concerns about the complexity of the structure, the feasibility of manufacturing it, and the ability to align all of the pieces. A design study is in progress to identify options for simplifying the design. One option involves incorporating the function of the inlet plenum within the design of the dump tank by adding a vertical pipe that joins to the two inlet pipes. Some consideration has also been given to making the dump tank section of the reactor vessel shorter in height and larger in diameter. An updated reactor vessel concept is shown in Figure 2.

- simplifying the chimney: The reference chimney is an hexagonal structure with a base plate that covered the top of the reactor vessel and welded gussets to stiffen the structure. This design makes the alignment of holes in the base plate with vertical penetrations in the reactor vessel more difficult and increases the potential for interferences between the gussets and irradiation devices installed in the reactor vessel. The design study is examining the possibility of a smaller diameter for the base plate and substituting stays for some gussets.

A method for providing local power control for the horizontal test sections has been studied. Two options were considered:

a) using soluble $^{10}B$ in a $D_2O$ annulus between the calandria tubes and the guard tubes (The guard tubes are part of the reactor vessel and maintain a flow of $D_2O$ around each horizontal test section after the $D_2O$ in the reflector vessel has been dumped) or

b) replacing the $CO_2$ gas in the annulus between the pressure tube and calandria tube with $^3He$ gas.

At this time, using soluble $^{10}B$ appears to more closely satisfy the users' requirements. The design study has not examined safety issues yet, and work on concepts for power control for the horizontal test sections is continuing.

A concept for a fuelling machine to handle the test fuel bundles and fuel channel components from the horizontal test sections is being developed. This work will provide a firmer cost estimate for the fuelling machine, will confirm the feasibility of using an adaptation of a CANDU fuelling machine and will define the space requirements within the Reactor Building for the fuelling machine and shielded enclosure. To guide the design study, a detailed definition of the requirements has been developed in consultation with the fuel development scientists and engineers:

- the handling of fuel bundles and fuel channel components must occur in a shielded enclosure at one end of the test sections because the access area to the test sections at the other end is shared by the beam users,

- the experimental fuel assemblies in the horizontal test sections may be shuffled between the three test sections as part of the experimental program, and

- provision is required for the removal of the assemblies to facilities remote from the IRF while maintaining a means of removing decay heat during the transfer.

A design study is also in progress to develop methods to transfer experimental fuel assemblies to and from the vertical test sections located in the reactor vessel at the bottom of the reactor pool while maintaining separation of the coolant from the pool water. This could be particularly challenging when organic coolants are used.

The concepts for the safety-related systems are being developed to provide information required by the safety analysis activities. The work is focused on the following systems:

- Reactor Regulating System (RRS): The CANDU software-based Distributed Control System and Plant Display System are being considered as the basis for implementing the RRS concept. In addition to providing monitoring and control for the reactor systems (e.g., primary cooling system, process water system and reflector cooling system), the RRS would also monitor and control the horizontal test facilities and the vertical test facilities. For neutron flux monitoring, the design study is assessing the use of fission chambers located outside of the reactor vessel and self-powered flux detectors located in tubes near the two core segments. The RRS concept relies on hafnium absorbers attached to drive mechanisms by electromagnets to control the reactor power. To support the development of the RRS concept, a control system modelling tool is being developed.

- Shut Down System One (SDS1): The CANDU SDS1 software-based trip system with two-out-of-three general coincidence logic for the reactor systems is being considered as the platform for SDS1 for the IRF. The design study is also considering grouped local coincidence logic for the experimental facilities (e.g., horizontal test facilities and the vertical test facilities). For neutron flux monitoring, the design study is assessing the use of fission chambers located outside of the reactor vessel and self-powered flux detectors located in tubes near the two core segments. The fission chambers will be "blind" to the neutron flux from the core when the $D_2O$ is dumped. SDS1 will shut down the reactor by de-energizing the electromagnets to disconnect the hafnium absorbers from their drive mechanisms. The design study is also assessing the possibility of implementing the logic to re-energize the electromagnets that connect the hafnium absorbers to their drive mechanisms using the RRS.

- Shut Down System Two (SDS2): The CANDU SDS2 software-based trip systems with two-out-of-three general coincidence logic for the reactor systems is being considered as the platform for SDS2 for the IRF. The design study is also considering grouped local coincidence logic for the experimental facilities (e.g., horizontal test facilities and the vertical test facilities). For neutron flux monitoring, the design study is assessing the use of ion chambers located outside of the reactor vessel and self-powered flux detectors located in tubes near the two core segments. The ion chambers will be "blind" to the neutron flux from the core when the $D_2O$ is dumped. SDS2 will shut down the reactor by dumping the $D_2O$ in the reflector. The design study is assessing the possibility of "poising" the $D_2O$ dump system using the RRS.

- Loops Emergency Cooling System: The horizontal test facilities and the vertical test facilities will operate at high temperature and high pressure. The design study is assessing the requirements for passive emergency cooling provisions to cover the transition from normal operating conditions to shutdown conditions, and to address upset conditions.

- Experimental Facilities Protection System: The design study is assessing the requirements for detecting and mitigating failures of the beam tubes, cold-neutron source system and neutron guides. Since the neutron guides will exit the Reactor Building and enter the Guide Hall, isolation valves will be required at the containment penetrations.

The following options for reducing the project costs are being studied:

- smaller diameter (i.e., 35 m rather than the reference 40 m) Reactor Building;
- smaller Operations and Utilities Building with a reduction in the number, size and scope of offices, laboratories and machine shops;
- incorporating the IRF mock-up rig within the Operations and Utilities Building;
- fewer support facilities for the neutron beam research program (e.g., reduced number, space and scope for offices, laboratories and machine shops);
- delaying installation of some experimental facilities (e.g., two rather than three loop systems for the horizontal test facilities, the BTF loop system, some neutron guides and two cold-neutron spectrometers);
- increasing reliance on existing facilities (e.g., machine shops, offices, hot cells and training facilities) and services (e.g., building heating and process water supply); and
- delaying construction of the Administration Building.

## 4. SUMMARY

An overview of AECL's case for replacing NRU with the IRF, and a summary of the reference IRF concept has been presented. The major activities in a pre-project program to reduce technical and licensing uncertainties, and to examine options for reducing the overall IRF project cost and schedule have been described.

## ACKNOWLEDGMENTS

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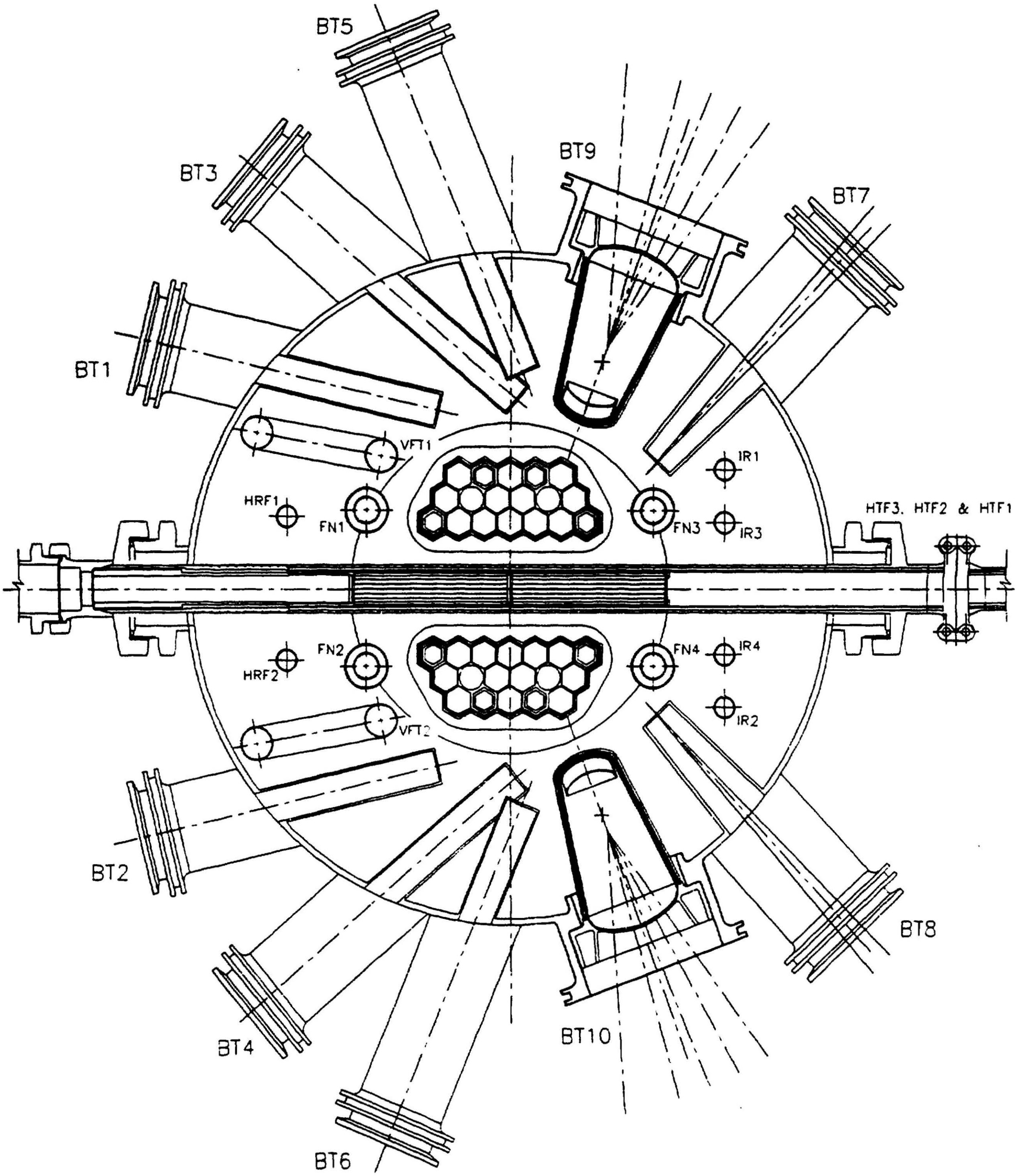


FIGURE 1. PLAN VIEW OF IRF REACTOR VESSEL

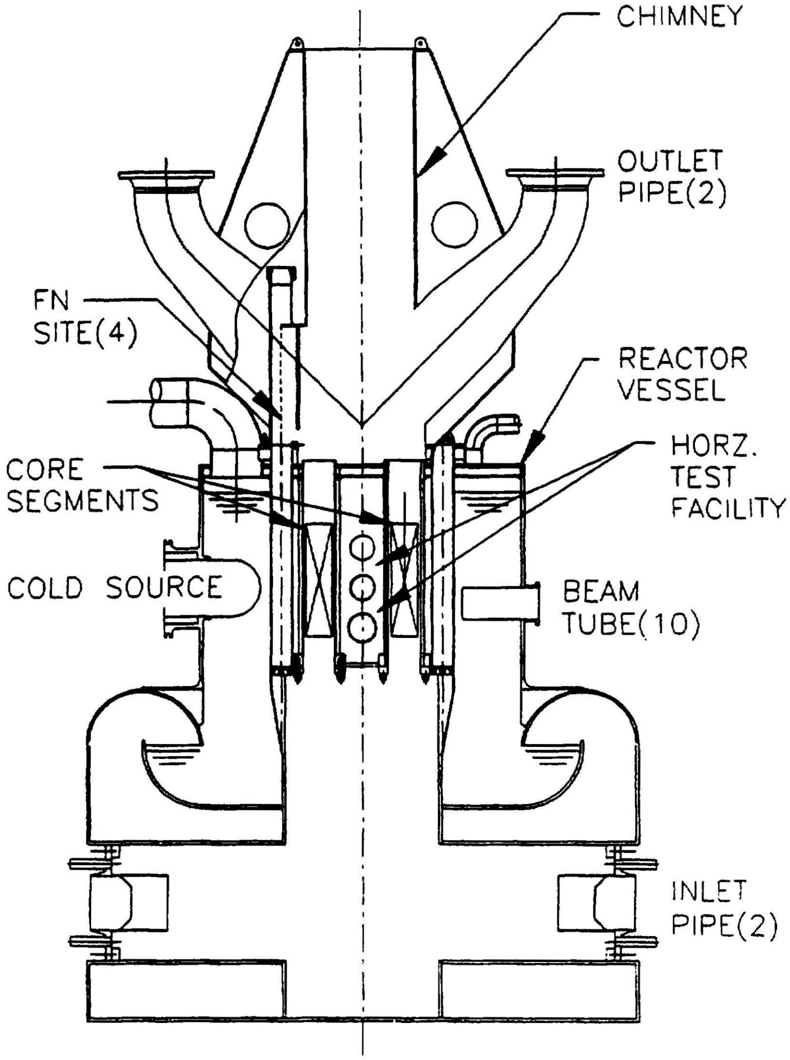


FIGURE 2. UPDATED REACTOR VESSEL CONCEPT