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MASTER

AN INTENSE STEADY STATE NEUTRON SOURCE - THE CNR REACTOR

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Introduction

The Center for Neutron Research (CNR) has been proposed in response to the needs - neutron flux, spectrum, and experimental facilities - that have been identified through workshops, studies, and discussions by the neutron-scattering, isotope, and materials irradiation research communities. The CNR is a major new experimental facility consisting of a reactor-based steady state neutron source of unprecedented flux, together with extensive facilities and instruments for neutron scattering, isotope production, materials irradiation, and other areas of research.

Center for Neutron Research

The major application for the new, ultrahigh-intensity steady state neutron source is in the field of neutron scattering, where the important applications in many areas lead to a large number of different experiments that justify the construction of a new, greatly superior source. The justification includes new experiments and new classes of materials, not accessible for research in existing facilities, as well as a substantial improvement in the quality of data that can be obtained from experiments that could be carried out, although less effectually, with presently available sources.

In considering the neutron source for the CNR, the first choice to be made is between a pulsed source and a steady state one. [Ralph Moon]

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The needs of users in the three major parts of the CNR research program - neutron scattering, isotope production, and materials irradiation testing - have been addressed by a series of studies, international workshops, and discussions.[1] The neutron flux and spectrum requirements as well as the size, type, and number of facilities required have been established quantitatively.

Table 1. User Needs - Neutron Scattering

Quantity	User needs
Peak thermal flux in reflector	$>5 \times 10^{19}\ m^{-2} \cdot s^{-1}$
Thermal/fast flux ratio at beam tube entrance	>80
Thermal flux at cold source position	$>2 \times 10^{19}\ m^{-2} \cdot s^{-1}$
Thermal and hot neutron beam tubes	16
Cold neutron guide tubes	8

Table 2. User Needs - Isotope Production

Quantity	User needs
Transuranium isotopes	
Epithermal flux at irradiation position	$\geq 0.6 \times 10^{19}\ m^{-2} \cdot s^{-1}$
Epithermal/thermal flux ratio	≥ 0.25
Positions	≥ 20
Available diameter of positions	≥ 16.6 mm
Available length of positions	889 mm
Other isotopes	
Thermal flux at irradiation position	$\geq 1.7 \times 10^{19}\ m^{-2} \cdot s^{-1}$
Positions	≥ 4
Diameter of positions	≥ 37 mm

Table 3. User Needs - Engineering Materials Irradiation

Quantity	Target region	Reflector RB positions
Fast flux	$\geq 1.4 \times 10^{19}$ $m^{-2} \cdot s^{-1}$	$\geq 0.5 \times 10^{19}$ $m^{-2} \cdot s^{-1}$
Fast/thermal flux ratio	≥ 0.5	≥ 0.3
Positions	≥ 6	≥ 8
Available diameter of positions	≥ 16.6 mm	≥ 48 mm

These requirements will be reviewed as the project progresses in order to be responsive to new scientific opportunities. The international scientific community will also have input into the design of experimental facilities.

In broad terms, the neutron scattering community requires the highest possible thermal flux in a large volume accessible to beam tubes and with space for hot and cold neutron sources. For isotopes production, a region of high epithermal flux is also needed, and for engineering materials irradiation a high fast flux zone. In addition, a minimum availability factor of 80-85% is desired.

In principle, such needs could be met either by a reactor or by an accelerator-based (spallation) source. The conclusion that a reactor neutron source can provide the desired flux, spectrum and reliability at a lower cost, and with less technical risk, was endorsed at a recent international workshop on advanced steady state neutron sources.[2]

A major decision that ORNL made concerning the reactor design is that the technical risk associated with this very advanced facility should be minimized, by basing the reactor as far as possible on known technology. Our main desire is to build a neutron source that can be used to do interesting and useful experimental research, not to do research on reactor design and construction (although some of that will be necessary and desirable). As we shall see, that decision led to the choice of a uranium silicide fuel form, in an aluminum matrix with aluminum alloy cladding.

Basic Reactor Design

The basic requirements of the core design are easy to see. The needs of the neutron scattering community dictate a high thermal flux *outsi e* the core, where there is good access and sufficient volume for a large number of beam tubes and sources. To produce many neutrons, a large number of fissions (i.e. a high power) and a core of small surface area are needed. In fact, for cores of basically similar composition and geometry, thermal flux in a reflector around the core is approximately proportional to neutron production or power level, and inversely proportional to the core surface area.

$$\begin{aligned}
\phi_{th} &\propto P/A \\
&\propto P/V^{2/3} \\
&\propto P^{1/3} \times (P/V)^{2/3} \\
&\propto P^{1/3}\,\bar{p}^{2/3}
\end{aligned}$$

A high power is beneficial, but a high power density is even more so.

In a conventional aluminum clad core, any one of several factors may limit the power density that can safely be attained. First, the small core volume limits the amount of fuel that can be contained and therefore the lifetime of the core. Second, the low volume limits the heat transfer surface available. The allowable heat flux is usually limited by one of three factors: by incipient boiling; by the bulk temperature at the core outlet (which we wish to keep below 100°C in order to avoid steam generation and possible explosion in the event of a loss-of-coolant-flow, loss-of-pressure accident); or by the fuel centerline temperature.

With reasonable coolant pressure and velocity, incipient boiling and bulk outlet temperature are usually the limiting factors for clean aluminum cladding. However, heated aluminum surfaces in water form an oxide layer. The oxide is in the form of boehmite, __________, which has a low thermal conductivity so that the temperature difference across the oxide under high heat flux conditions leads to an unacceptably high fuel centerline temperature.

In order to carry enough fuel in the core to have a long enough fuel cycle, a high density fuel form is needed. Based on the work of the DOE's Reduced Enrichment for Research and Test Reactors (RERTR) program, an obvious choice is one of the silicides (Fig. 1). In-pile tests[3] have shown the U_3Si_2 form to be very resistant to swelling and other undesirable effects during irradiation, which the U_3Si_2 compound is not. The U_3Si_2 compound is the one selected by the RERTR program for use at high loadings for reduced enrichment fuel plates, and is currently being demonstrated as a full-scale core in the Oak Ridge Research Reactor.[4]

In a high power density, and therefore high heat flux, it is essential that the fuel have a high thermal conductivity even when a heavy loading is used. The problem does not arise in conventional research reactors, and indeed for low fuel loadings the thermal conductivity for all fuel forms approaches that of the aluminum matrix. As Fig. 2 shows, at high uranium loadings (3 to 4 g/mL is the minimum required for a reactor of this kind) the silicide offers a very great improvement over the oxide and aluminide forms used in most research reactors.

These thermal-hydraulic constraints are collected in Fig. 3, which clearly shows the effect of the oxide layer in limiting the attainable power density. At a D_2O coolant velocity of, say, 27 m/s a fuel plate containing 3.3 kg of uranium per liter of U_3Si_2/Al disperse and with a 30 μm oxide layer (about 14 days accumulation under these circumstances) on the cladding is limited to a maximum hot spot power density of 7.9 MW/L before exceeding the permissible fuel centerline temperature. In a similar plate without any oxide, the fuel centerline temperature never approaches its limit, and a hot spot power density of 16 MW/L can be accommodated before incipient boiling or before the bulk outlet temperature reaches 100°C. Using the relationship described earlier among power, power density, and flux, we see that an oxide-free core can give approximately twice the flux available from one with a normal oxide layer.

Calculations show that even with the anticipated oxide growth on untreated surfaces, a core can be designed to meet the criteria listed in Tables 1, 2, and 3. However, suppressing the oxide growth would lead to an even more exciting facility, with a peak thermal flux of 10^{20} neutrons $m^{-2} \cdot s^{-1}$.

In principle, there are several ways in which oxide growth might be avoided - for example by changing to a stainless steel or zircalloy clad, by treating the surface of the aluminum to form a corrosion resistant layer, or by adjusting the water chemistry. In practice, the proposed water chemistry may already be close to optimum from this point of view, and both zircalloy and stainless steel are much more expensive as a cladding material than aluminum alloy; in addition, stainless steel absorbs enough thermal neutrons to affect the flux and reactivity adversely. A surface treatment by ion implantation or ion mixing seems to be the most promising approach, and initial estimates are that the treatment would add only 10-15% to the reactor operating costs, compared with untreated fuel elements, while offering the potential for a two-fold increase in flux.

Figure 4 summarizes the constraints. It includes allowances for the difference between peak power density, which determines the local hot spot conditions, and average power density, which determines the flux. For a very high power density, the core volume is so small that there is not enough fuel, even using a high density silicide, to provide a two-week core life at a reasonable burnup (say 40%). The hatched area indicates the region in which the constraints and the design criterion for thermal flux are simultaneously satisfied. Two reference core designs, discussed in the next section, are shown; one is for a core with a typical oxide layer and the other for a core in which a major reduction in oxide growth rate has been achieved.

Example Core Concept

As a reference design, ORNL has chosen an annular core with aluminum clad, silicide fuel. The overall concept is very similar to the HFIR, which has a long record of very successful operation with an availability of 90 to 95%. The core, chosen after extensive analysis and optimization of the neutronics and core physics,[5] consists of two concentric annuli, each made up of involute fuel plates: thus, only two different plate types are involved in the entire core. Between the inner and outer annuli is a 25-mm thick annular aluminum ring with axial holes forming irradiation positions. The overall length is 450 mm, including a 50-mm long unfueled region at each end of the core. There may be some advantage to dividing the core into an upper and lower half,[6] with a space in-between for remixing of the coolant flow; beam tubes pointing at the space between the two core halves may gather a higher ratio of thermal-to-fast neutrons than beam tubes opposite a fuel element. However, the horizontally-divided core would probably be somewhat less neutronically efficient than a single core, requiring a higher power to produce the same flux. No detailed comparison of the single and the divided core based on a common set of assumptions and correlations has yet been made, but it will be, and the actual core of the CNR reactor when it is built will adopt the most effective and efficient design that can be desired.

The core is surrounded and moderated by heavy water. Because of the relatively poor moderating properties of D_2O, there is relatively little moderation within the fueled region of the core. The spectrum in the irradiation facilities in between the two fuel annuli is therefore rich in fast and epithermal neutrons, but with only a small thermal component (Fig. 6), making these positions highly desirable for engineering materials irradiation experiments. Just outside the fuel is a region of very high epithermal flux, well suited to the production of the trans-uranic isotopes, and further out still a very broad peak of extremely thermal flux. The high thermal flux zone contains the beam tube mouths, facilities for isotopes production and one or more cold sources (perhaps liquid deuterium) to provide a high flux of very low energy neutrons for scattering experiments. There will also be a hot source, probably a block of graphite heated to 200-2500 K (Fig. 7).

The high thermal flux, and the very large volume of the high thermal flux region, is indicated by Fig. 8, which compares the CNR reactor with two important existing high flux reactors. Notice that the thermal peak in the HFIR's beryllium reflector is much narrower than that in the D_2O reflectors of the CNR or of the Institut Laue Langevin (ILL) reactor.

Two curves are shown for the CNR, one for a power level of BSMW and one for operation at 270 MW. The active core volume is 35 liters, so the two power levels correspond to an average power density of 3.9 MW/L and 7.7 MW/L, respectively. The former is believed achievable even in the presence of 14 day's growth of oxide on the aluminum cladding, the latter requires an 80-90% reduction in the oxide formation rate.

In order to achieve an acceptable ratio of the peak-to-average power densities in the core throughout the cycle, the fuel loading is graded both radially, as it is in the HFIR fuel plates, and axially; see Fig. 9. All these calculations - including those for the ILL and HFIR reactors with which comparison is made - correspond to the unperturbed flux at the end of the cycle, i.e. the flux in the absence of beam tubes, irradiation experiments, or control rod effects. Future work will, of course, take these perturbations into account.

The major assumptions, cross section sets, and codes used in the physics analysis are shown in Table 4: This information is provided to facilitate comparisons and cross checking of our calculations by other groups. Similarly, Table 5 lists the data sources and correlations used in our thermal-hydraulic calculations.

Table 6 lists the major specifications of the reference core for the two power levels chosen as an example (135 MW and 270 MW). Table 7 shows the results of the calculated parameters of the reference core compared with the objectives set by the research community for the new neutron source. Even at the lower power level - referred to as the baseline case in Table 4 - the criteria are met and, in most cases, very substantially exceeded. The higher power version, dependent on improvements in oxide control, looks even better of course.

Safety

Up to now (May 1986) the major part of our effort has been devoted, as described above, on establishing the main parameters and design features of a reactor that will meet the requirements of the research communities who will use it. That point has recently been reached, and it is now possible to begin work on safety analyses, and on planning the R&D and other work necessary to establish the safety of the proposed designs. Compared with a PWR power reactor, the CNR (like the very similar HFIR, which has operated safety for ? years) has a very small fission product inventory, decay heat, and radioactive waste production rate, although its decay heat power density is high. Similarly, the weapons grade material inventory is very small (18 kg of U-235 compared with 280 kg of plutonium in the spent fuel assemblies discharged at the end of each PWR fuel cycle). However, in an unirradiated research reactor core, the material is relatively accessible; in spent fuel it is available only following hazardous chemical processing.

With regard to the safe operation of the reactor under normal conditions, careful and thoughtful attention must be paid to a large number of issues, including the effects of voids, by loss of pressure (which is why we have chosen to limit the bulk outlet temperature to less than 100°C) following a small pipe break, loss of pumping power and so on. All of this, and more, is part of the planned program relating to safety and environmental issues.

Summary

A new research reactor to serve as a neutron source for materials science and isotopes production is being planned. In order to minimize the technical risk of the project, a fairly conventional aluminum clad core has been chosen. The coolant, moderator, and reflector are heavy water. By using the recently developed uranium silicide fuels, rather than the oxide used in most older research reactors, a very high power density core can be designed while maintaining an acceptable core lifetime. Extensive analysis has led to an initial choice of a core that is basically very similar to the extremely successful HFIR core. The calculated thermal flux in the reflector is several times higher than in any existing reactor. With a credible reference core selected, safety analyses have just begun.

Table 5. Thermal-hydraulic design assumptions and correlations

Factor or issue	Assumptions or correlations
Safety margins	Deterministic; estimated engineering hot spot factors are included as well as calculated nuclear hot spot factors[a]
Oxide growth on cladding	Griess correlation (extrapolated). Oxide thickness. ∂_{ox} reduced by 50% to approximate moving hot spot. $(\partial_{ox})_{max}$ = 0.031 mm (∂_{ox}) initial $\leq$ 0.01 μm
Oxide thermal conductivity	2.25 W/mK mean
Maximum design inlet D_2O temperature	49°C. Based on a generic heat rejection capability.
Maximum allowable outlet D_2O temperature	99°C. Based on avoidance of steam generation in the event of a small-pipe-break LOCA.
Maximum fuel temperature	Based on ORNL fuel-core, cladding, and oxide thermal conductivity data. Taken to be 354°C for U_3Si_2 at the hot spot.
Calculation of critical velocity (for plate collapse)	Finite element analysis. $V_c \approx 78.5$ m/s at 316°C.
Maximum value of heat flux divided by incipient boiling heat flux	1.0
Minimum acceptable core lifetime	14 d at full power
Probable fuel-plate cladding material	Al alloy 6061, probably with surface treatment
Correlation for friction factor	Filonenko (for non boiling turbulent flow)
Correlation for heat transfer coefficient	Petukhov (for non boiling turbulent flow)
Correlation for incipient boiling heat flux	Bergles-Rohsenow (combined with correlation of previous item)
Correlation for critical heat flux	Gambill (superposition) - updated
Coolant outlet pressure	4.1 MPa
Thermal analysis computer code	Code previously developed for HFIR now modified to use D_2O property values.

[a] For example, a hot spot maximum power density of twice the average power density, (including a calculated nuclear peak-to-average ratio of 1.60 in the present core design).

DISCLAIMER