Options for the Delft Advanced Neutron Source

H.P.M. GIBCUS[1], P.F.A. DE LEEGE[1], F. LABOHM[1], J.W. DE VRIES[1], A.H.M. VERKOOIJEN[1], J. VALKO[2], W. FELTES[3] and J. HEINECKE[3]

ABSTRACT

Results of feasibility studies are presented for options for an advanced neutron source for the Delft reactor including upgrading the HOR, a 2 MW pool-type research reactor at the Delft University of Technology. The primary utilisation of the HOR focuses on beam research applications with neutrons and positrons. The aim of being scientifically competitive in that research area requires a thermal neutron flux level of at least 1×10^{14} n/cm^2/s. The feasibility of an accelerator driven neutron source and upgrading the present core to a super compact core for reaching this goal has been investigated at large from a safety and operational point of view. For the upgraded core, a 3x3 fuel assembly arrangement and beryllium reflected at all sides was chosen. Figures on the system performance, including the merits of a cold neutron source application feeding the neutron guide system, are presented.

1. Introduction

IRI employees carry out application-oriented research with or on ionising radiation. This research is strongly experimental in nature and for this the institute has, for a long time, had a nuclear reactor. This should mainly be seen as the radiation source that provides a number of instruments with neutrons and positrons. In addition to the reactor, the institute has a number of other experimental facilities in which various forms of ionising radiation are used.

A number of facilities are of importance on a world scale. In particular the intense positron bundle (POSH), the electron accelerator with the time-resolved microwave absorption set-up and the instruments that use polarised neutron bundles.
At the moment, the reactor plays an important role in the following areas:
- neutron bundle research and the development of new instruments;
- research with the new POSH bundle of low energy positrons;
- preparation of most short-lived isotopes and the execution of INAA.

The IRI has considered the question of how the content of its research should develop and what technical facilities the institute should have available in a 5-15 year period.
Within the framework of the IRI strategy, an analysis has been made of the scientific fields in which important developments can be expected:
From these we have chosen as its spearheads: materials, sensors and instrumentation, energy and sustainable production, environment and health
Three possible alternatives have been developed for the infrastructure necessary to reach the research objectives:
- An institute that "upgrades" the existing reactor;
- An institute that uses (a number of) dedicated accelerators;
- An institute that only uses external facilities.

It is clear that it is possible to continue doing research within the chosen themes and at a high scientific level if the HOR were to be shut down and if there were no larger accelerators available. There would however be a significant shift in the subjects being researched. A number of research projects that appear to be very promising will certainly no longer be possible. Also in this situation, significant budgets, both initial and annual, must remain available to gain access to large external facilities and for the purchase of isotopes. Not having a large neutron source and experimental facilities available where master and PhD students can learn their experimental skills and to which researchers have low threshold access is experienced as a negative aspect.
A last negative aspect is the loss of the cohesion of the institute and the role it plays as a knowledge centre for the use of neutrons and other ionising radiation.

Presenter: Adrian H.M. Verkooijen,
[1] Delft University of Technology, Interfaculty Reactor Institute, Mekelweg 15, 2629 JB Delft The Netherlands
[2] Ikesol Consulting, Rooseveltlaan 6, NL 2625 GN Delft – The Netherlands
[3] Framatome ANP GmbH, P.O. Box 3220, 91050 Erlangen – Germany

For the accelerator-based system, studies were initiated and have been going on in the USA for a linear accelerator (LINAC) type system of moderate power called LENS at Indiana University. This design uses a pulsed proton beam for generating neutrons by hitting an appropriate target. The results of these studies have been used to perform calculations for a LINAC system at IRI on the neutron performance of such a system, using a CNS for feeding the neutron guide system. For a High Current LINAC (50 mA peak current, duty factor 10%, beam pulse length 450 ms) with 12 MeV H^+ beam and Be-target for the production of pulsed neutrons the following performance may be expected. The time averaged cold neutron flux available at the neutron guides will have an integral value of 7.5×10^6 $cm^{-2}.s^{-1}$. This value can be compared with that supplied by the upgraded reactor to which the remaining of this article is devoted.

2. Introduction HOR

The HOR is a pool-type research reactor at the Interfaculty Reactor Institute (IRI) of the Delft University of Technology using MTR-type fuel assemblies. It has been in operation since 1963, maintaining good performance by upgrading instruments and beam tube facilities as well as keeping good maintenance records over the years for the reactor equipment and ancillary facilities. The reactor core set-up itself has been modified several times, the latest major change being the core conversion form HEU to LEU fuel with core compaction from 32 to 20 fuel assemblies. At the moment, the HOR is operated on a regular basis at a thermal power level of 2 MW, attaining a modest thermal flux of about 2×10^{13} n/cm^2/s.

An analysis of the performance of operating research reactors and their utilisation for neutron beam research clearly shows that to be scientifically competitive in that research area, such facilities should at least exceed an unperturbed thermal neutron flux level of 1×10^{14} n/cm^2/s at the thermal flux peak. The maximum flux should occur at the position of the beam port entrance coupled with a cold neutron source. Feasibility studies have been performed for modifying and upgrading the HOR with the aim to improve the utilisation performance in such a way that the above mentioned goal can be reached, without exceeding the maximum (for the time being) licensed thermal power level of 3 MW. These studies follow an integrated approach and are pertaining to the core itself and the beam port facilities, including a cold neutron source, neutron guides and coupling of core and facilities. Further studies are ongoing to improve the relatively short operation cycle and the relatively low burn-up figures, which are disadvantageous economically. Thus, preliminary results on improvement of the operation cycle and fuel management when using higher fuel loadings are reported as well.

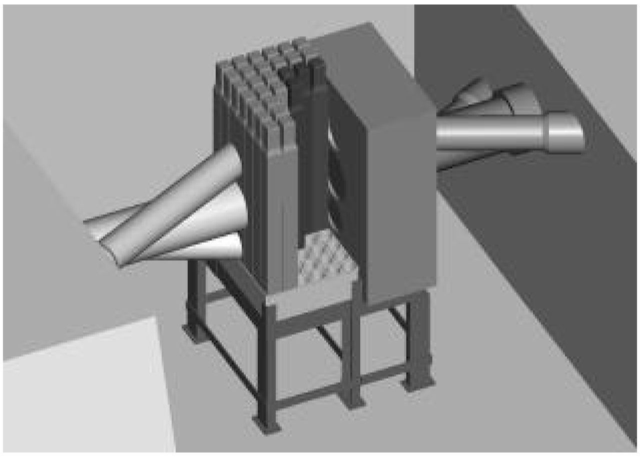

Fig. 1. General view with partially loaded core
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Fig. 1 gives a general view of the proposed core and beam arrangement showing the core only partially loaded with fuel and reflector assemblies. The assumed HOR core is a 3x3-arrangement, using LEU silicide fuel at the highest licensed and industrially available U-density of 4.8 g/cm^3. The core is composed of 5 standard fuel assemblies with an initial ^{235}U-loading of 420 g each and 4 control fuel assemblies with an initial ^{235}U-loading of 310 g each. The core is beryllium reflected at all sides, except at top and bottom. Moreover, at one core side the beam ports, including a cold neutron source facility, are embedded in a beryllium reflector block. A core shroud to have clear coolant flow conditions surrounds the entire region of core and reflector, which consists of single assemblies. This shroud may make a new grid plate necessary.

In Section 2, a comprehensive screening study [1] for a very substantial core size reduction and its consequences are discussed. In Section 3, the application of an elaborate MCNP model for calculating HOR configurations and fuel cycle issues is highlighted. Section 4 discusses the performance gain at the experimental facilities for different sets of modifications. In Section 5 some conclusions are drawn.

3. Screening study

In the screening study, the feasibility of a super compact core with only 9 fuel assemblies was investigated by performing neutronic and thermal-hydraulic calculations for 3 MW power level operation. The operational conditions were evaluated at the same time. The screening study was split into two parts, a nuclear and a related thermal-hydraulic analysis of the compact core plus of some accidental conditions, which may limit the compacting.

3.1 Nuclear analysis

Step 1 of the nuclear analysis was performed by applying a burn-up chain with xy-diffusion code for each burn-up step of the core set-up (Fig. 2) on basis of the ENDF/B-IV cross section library. Group cross sections were established by transport code generated flux weighing.

Step 2 of this analysis applied MC-code MORSE-K to check and to adjust the results of step 1 at BOC by these high accuracy calculations. These calculations were performed in 60 energy groups based on the JEF-1 library. Special emphasis was put on the exact modelling of the geometry of core, reflector, supporting structures and the rather wide beam tubes. Shut down margins were also received from MC-calculations. The actual HOR-license requires the assumption of 2 out of 4 shut down rods stuck as worst case.

Two approaches were made for the fuel management: Exchange of 1 fuel assembly or of 2 per cycle. Although the exchange of 2 per cycle need higher excess reactivity at BOC, the stuck rod case of 2 units simultaneously could be met. A higher U-235 loading however than what is possible today (4.8 gU/cm^3) may need to limit the management to the exchange of 1 assembly per cycle. With the actual foreseen fuel using the currently available U-density the two cycles under consideration lasted 26 FPD (1 assembly exchange) and 48 FPD (2 assemblies exchange), respectively. These cycles which are relatively short for a 3 MW research reactor, are partially also a consequence of the loading scheme, which is entirely directed to enhance the flux in front of the beam tubes, especially at the later position of a cold neutron source (see Fig. 2). The calculations of the 3x3 core demonstrated an adequate safety margin to exist for shutting down the core at any state during the operation cycle.

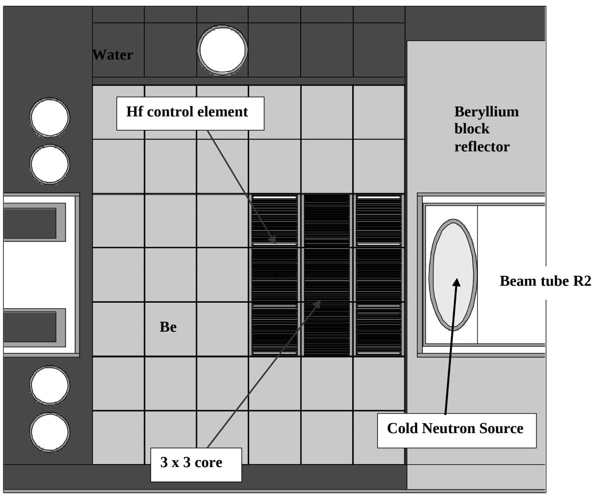

Fig. 2. Layout of 9-element core

3.2 Thermal-hydraulics analysis

The enhancement of the performance of the HOR is linked to a higher value of the average power density in the core, which is mostly originating from the compacting of the core. That compacting cannot be recommended without a check of the heat removal capacity of the HOR-plant and of the thermal-hydraulics design of the new HOR-core.

As the HOR is a low-pressure system, the power density of the reactor core is limited by the requirement that hydro-dynamically stable coolant flow must be ensured in all cooling channels of the core. A violation of this criterion would lead to a drastic deterioration of the heat transfer accompanied by a sudden rise of the fuel plate temperature to an intolerable level.

The screening of the thermal-hydraulics design deals with the determination of a) the coolant flow rates through the various coolant channels of the core based on an overall flow rate at HOR of 80 kg/s, b) the corresponding pressure losses, c) the temperatures in the coolant and in the fuel plates, and d) the safety margins against the above mentioned flow instabilities.

The applied code system for thermal-hydraulic design of research reactors basically consists of a survey code for parametric studies, a network code to determine flow distributions within a network of hydraulic resistances and a sub-channel code to investigate the thermal-hydraulic details within the sub-channels and the fuel plates of the fuel assemblies.

The maximum permissible power, which can be extracted from a cooling channel at low system pressure under forced convection conditions, is limited by the excursive flow instability (FI) phenomenon. The numeric quantity suitable to comprehend the FI-phenomenon is the flow instability power ratio FIPR describing the factor by which the reactor power can be increased without violation of the criterion of the minimum acceptable safety margin.

In case of the HOR compact core, about 82% of the total flow rate is used to cool the fuel plates, which is quite a good value expressing a sound design. The irreversible pressure loss across the core amounts to 0.138 bar, larger than the value of 0.026 bar of the present core. Having however in mind that the pump head at nominal flow rate of 80 kg/s is 3.2 bar, this increase of the flow resistance of 0.112 bar seems to be of minor importance.

For the HOR compact core the analysis of the coolant channels (10 different types) resulted in a maximum meat temperature in the centre of the plate of the hottest channel of 91.2°C. The minimum FIPR of 2.42 for the related sub-channel indicates that the HOR 3x3 core is well protected against instabilities; the well-known minimum bubble detachment parameter η linked hereto is 179.1 cm^3 K/J.
In the limited transient analysis as performed for the screening, the Loss of Flow-accident (LOF) turned out to be the limiting case. Thus, it was decided to consider only the LOF-accident within the screening. The main result of the LOF-analysis is documented in Fig. 3.

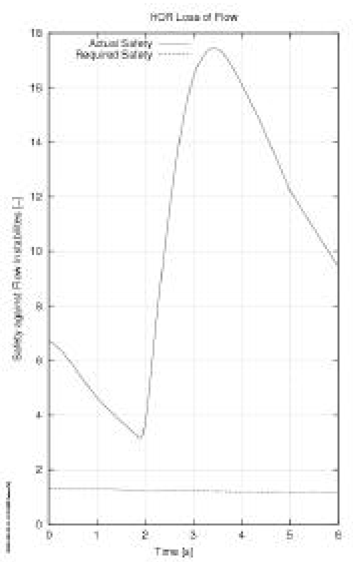

Fig. 3. Safety at Loss-of-Flow

During LOF the forced convection mode with top to bottom flow reverses into the natural convection. During that reversal the axially averaged heat flux of the hottest plate is about 4 W/cm^2. This value is well below 6 W/cm^2, which is the limiting value, which guarantees that pressure pulses during the chugging of the coolant are less than $\pm$ 0.1 bar in order to avoid additional mechanical loads to the plates. The value of 6 W/cm^2 has been determined experimentally in a test section with flow reversal simulation. The LOF-analysis resulted in a safety against flow instabilities of 3.17.

It is obvious from the results that there is quite a large flexibility with respect to the pump half time, as the safety against flow instabilities gained from the screening is larger for all considered cases than that minimum acceptable safety of 1.27. From the analysis, it follows that the thermal-hydraulic safety margin for operation as well as for the design basis accidents to be assumed was demonstrated to be quite sufficient.

4. MCNP model calculations and fuel cycle issues

As mentioned before the screening study resulted in comfortable safety performance figures for the adopted fuel design and power level. On the other hand, the operation cycle length turned out to be relatively short for a 3 MW reactor, also resulting in relatively low burn-up figures for End-of-Life-fuel (EOL). To improve the situation, one could try to go to higher U-densities in the meat. However, higher density fuel than the present limit of 4.8 gU/cm^3 is not expected to be licensed and commercially available shortly.

Another possibility for increasing the assembly loading is by changing the fuel geometry using thicker fuel plates with thicker meat [2]. Of course, at the same time the number of fuel plates of the FA decreases, but if the thermal-hydraulic safety margins are sufficient still, this could be a good option. As has been demonstrated by the screening study, this is a valid proposition for the HOR case studied.

A general investigation has been performed concerning the merits of such a choice to get an impression of possible operation cycle improvements, without going into optimisation and additional safety assessment considerations. The technical details in terms of the calculation approach for the present HOR configuration and for two possible fuel options, indicated as the HOR-1 and HOR-2 cases are discussed below. Furthermore, the calculation model includes a Cold Neutron Source option for considering the utilisation aspects for different fuel assemblies and core set-ups.

4.1 The MCNP model of the HOR

Independent of the work under 2.1 above, a MCNP model was established to calculate present HOR configurations and to study the upgrading alternatives. The model describes the fuel assemblies by individual fuel plates and uses 15 axial regions along the height of the fuel. Each movable beryllium reflector element is modelled separately. For the upgrading studies an optional beryllium block reflector of 70×30×100 cm is also included. All the existing beam tubes are represented in the model. The most important beam tube R2 is modelled including the concept cold neutron source: an ellipsoidal region of hydrogen in aluminium cladding. In x-y-z directions the overall 3D model extends to 180×155×180 cm. The insertion of the control absorbers can be changed individually.
The material composition of the fuel is given per fuel assembly and per axial region. The burn-up dependent fuel composition is determined separately. A modified scale sequence called SAS6 is used to generate on cell level detailed nuclide densities for different burn-up levels. The generated tables can be used in different codes like Bold-Venture, KENO and MCNP. In this case, the excess reactivity (xenon-free, cold condition) for fresh fuel and at different stages of average burn-up (uniform for all assemblies) has been determined. However, an axial burn-up shape is used for each assembly, validated for the HOR configuration and determined from nodal 3-D calculations. The depletion of beryllium and the photo neutron production in beryllium have been neglected. For the calculations the ENDF/B-VI data library was used as distributed with the MCNP4C-2 package. The model has been validated on a recent HOR configuration by comparing reactivity and the measured axial distribution of ^{59}Co activation reaction rates. The average value of the ^{59}Co activation measurement was used for power normalization, the ratio of the total power to fission power is assumed to be the same at all systems. A horizontal cross section of the 9-element core as plotted by MCNP is shown in Fig. 2.

4.2 Calculations with MCNP

Calculations have been performed with two types of (conceptual) fuel: HOR-1 and HOR-2. The HOR-1 fuel type is nearly identical to the fuel characteristics used in the screening study discussed in Section 2, whereas the HOR-2 fuel loading is taken from one of the cases considered in [2]. The relevant data characterizing these fuels are given in Table 1. Just for reference, the fuel characteristics of the present HOR configuration are also listed in Table 1.

Table 1. Characteristic data of two potential fuels compared to the present HOR fuel

	HOR	HOR-1	HOR-2
Fuel meat material	U_3Si_2-Al	U_3Si_2-Al	U_3Si_2-Al
Uranium density [g/cm^3]	4.3	4.8	4.8
^{235}U loading per standard FA [g]	300	410	546
Enrichment [%]	19.75	19.75	19.75
Meat thickness [mm]	0.5	0.51	0.76
Cladding thickness [mm]	0.35	0.39	0.38
Fuel plate thickness [mm]	1.20	1.29	1.52
Moderator channel thickness [mm]	3.0	2.23	2.46
Hf absorber blade thickness [mm]	n.a.	3.1	3.1
Number of plates in standard FA	19	23	20
Number of plates in control FA	10	17	15

For each configuration the critical rod position, the shutdown reactivity and the excess reactivity were calculated. In addition, the reactivity when any two control rods are in the fully withdrawn position while the other two are fully inserted, was also determined (see para 2.1). The shutdown conditions have been met for in all configurations.

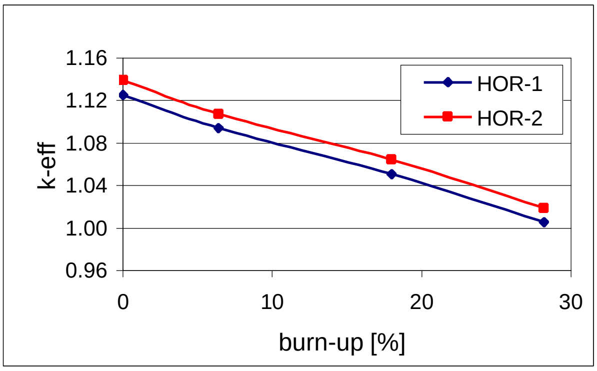

Fig. 4. k_{eff} as function of uniform burn up

A set of calculations was performed with uniform burn-up, i.e. where all fuel assemblies in the core have the same burn-up. The dependence of k_{eff} on uniform burn-up for the two types of fuel is shown in Fig. 4. For purposes of comparison BOC- and EOC-excess reactivity values were defined. Calculations with a realistic burn-up distribution and with all control rods fully withdrawn, gave k_{eff} (BOC) = 1.068 and k_{eff} (EOC) = 1.043 for cold, clean core conditions. The same values were also achieved by applying uniform (average) burn-up levels. For HOR-1 the average burn-up at BOC was 13.4%, the EOC value was 19.8%. For HOR-2 17.0% at BOC and 22.8% at EOC are the corresponding figures. These values, together with the resulting cycle data are displayed in Table 2. It can be seen that although the difference in burn-up at BOC and EOC is less for HOR-2 than for HOR-1, still, due to the much higher ^{235}U loading of the HOR-2 fuel, the achievable cycle is substantially longer. Due to the simplification with uniform burn-up, in practice the achievable operation cycle will be somewhat shorter than the figures displayed in Table 2. For actual burn-up distributions, the figures are estimated to be lower by about 15%.

Table 2. Cycle length calculation results for HOR-1 and HOR-2 fuel options

		Burn-up [%]		^{235}U consumed [g]		Cycle length [MWd]	
	k_{eff}	HOR-1	HOR-2	HOR-1	HOR-2	HOR-1	HOR-2
BOC	1.068	13.4	17.0	208	254	166	203
EOC	1.043	19.8	22.8				

5. Performance

To gain an insight into the improvements of the performance by upgrading the HOR the neutron spectra at the experimental facilities were calculated. In the summer of 2000, at the HOR a multi-beam neutron guide system (NGS) was installed in beam tube R2 [3]. This NGS consists of four neutron guides, delivering neutron beams with high signal-to-noise ratio at the instruments in the experimental hall. Further the NGS is designed in such a way as to facilitate, in a later stage, the incorporation of a CNS. Beam tube R2 is considered to be the most important beam tube with respect to neutron beam research at IRI. Thus, performance calculations were restricted to this beam tube.
The calculation of the neutron flux at the experimental facilities was performed in two steps. First the wavelength-dependent flux for neutrons travelling in the direction of the neutron guides was calculated, both at the surface of the reactor core and at the surface of the CNS. A 42-point energy distribution and four direction values were used.

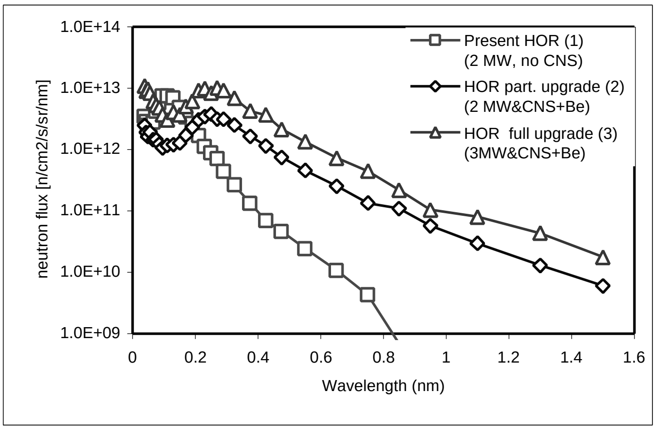

Fig. 5. Cold Neutron Source exit spectra

Secondly the wavelength-dependent neutron fluxes at the exits of the neutron guides were determined.

The wavelength-dependent transmission of the NGS was determined by means of a Monte-Carlo code. This code treats the horizontal and vertical phase space separately, so the round surfaces of the CNS and the experimental tube R2 were approximated by square surfaces of the same area. As the outer diameter of the CNS is smaller than the inner diameter of R2, part of the neutrons entering the NGS do not come from the CNS but directly from the surface of the reactor core. Therefore three calculations were made: the transmission of a) neutrons coming from the CNS (A), b) neutrons coming directly from the core for an area equal to the inner area of R2 (B), and c) neutrons from the core but for an area equal to that of the CNS (C). The transmission of the NGS was then given by A+B-C.

This method was used before and has proven to yield neutron fluxes, which are in good agreement with, measured values [3,4]. Three different cases have been considered, involving differences in system configuration and operating power level:

1) present compact core configuration with 20 fuel assemblies and beryllium reflected on three sides at 2 MW power;
2) present compact core configuration, beryllium reflected at all sides and CNS installed in beam tube R2 and power level of 2 MW (partial upgrade case);
3) super compact core configuration with 9 fuel assemblies, beryllium reflected at all sides, CNS and increased power level of 3 MW (full upgrade case).

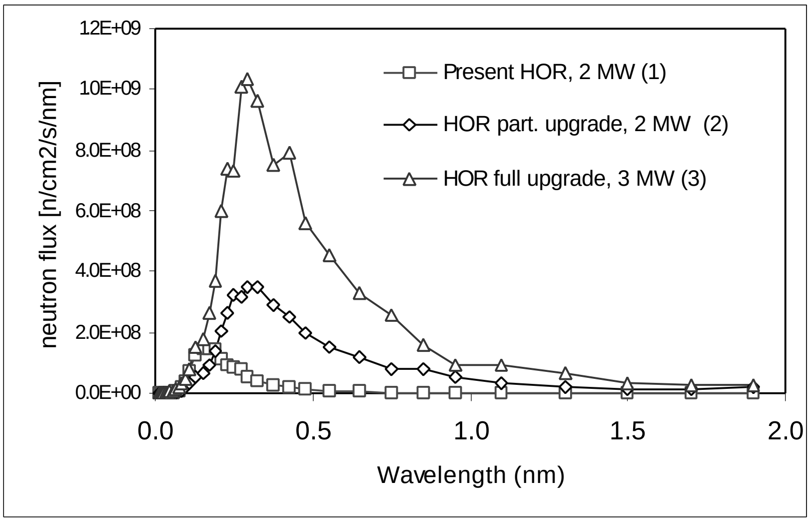

Fig. 6. Cold neutron flux at neutron guide system exit

Fig. 5 and Fig. 6 show the results of the performance calculations. For the fully upgraded HOR, the cold neutron flux at the neutron guides exit is calculated to an integral value of 4.4×10^8 n/cm^2/s, which is comparable to the figures achieved at the Swiss continuous spallation source SINQ and a research reactor like the FRG-1. Another common figure-of-merit in connection with neutron beam research is given by the neutron flux per unit wavelength at a wavelength of 0.45 nm. With a value of 6.8×10^8 n/cm^2/s/nm for the upgraded HOR, this is about a factor of 40 higher than the present performance.
Table 3 lists both performance figures for the three cases considered.

Table 3. Performance figures for different core design options

case number	1	2	3
Integral cold neutron flux [n/cm^2/s]	2.9×10^7	1.6×10^8	4.4×10^8
Neutron flux at 0.45 nm [n/cm^2/s/nm]	1.6×10^7	2.2×10^8	6.8×10^8

Naturally, the partial upgrade option results in lower performance figures than the fully upgraded case, but it may be still an interesting proposition from a cost-benefit point of view.

In figure 7 the results of the accelerator calculations for the cold neutron flux at neutron guide system exit are compared with those of the upgraded reactor.

5. Conclusions

The feasibility of upgrading the HOR to a scientifically competitive level of performance has been investigated at large. The work scope included a screening study, considering the nuclear and thermal-hydraulic aspects of a super compact core design of a 3x3 fuel assembly arrangement. Independent of the screening study, an MCNP model was set up for studying the upgrading alternatives. The studies included fuel cycle issues and the performance of a Cold Neutron Source option. Also, calculations were performed concerning the wavelength-dependent neutron intensity at the exit of the multi-beam neutron guide system.

The screening study indicated an adequate safety margin to exist for shutting down the core at any state during the operation cycle under the assumption of 2 out of 4 shut down rods stuck as worst case. The operating cycle was investigated on the basis of the exchange of either one fuel assembly per cycle or two assemblies per cycle. The resulting cycle length for both options turned out to be rather short for a 3 MW research reactor. This is partially due to the loading scheme, which is entirely directed to enhance the flux in front of the beam tubes and at the cold neutron source position.

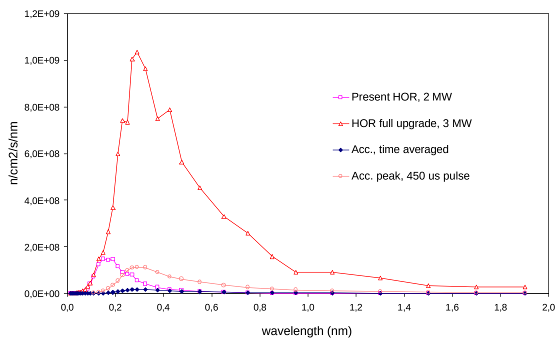

Fig. 7.
Cold neutron flux at neutron guide system exit

In a separate study the merits of a different fuel design with higher fuel loading of 546 g ^{235}U per fuel assembly (density 4.8 gU/cm^3) for increasing the operation cycle length and discharge burn-ups were investigated. It seems feasibly to increase the cycle length by about 20 % in comparison to the fuel design adopted earlier, resulting in substantially higher discharge burn-up figures. Further optimisation and safety margins for higher loadings have not yet been considered by now.

The screening of the thermal-hydraulics of the shrouded core design resulted in adequate margins against flow instability for 3 MW operation and for the investigated LOF-accident for the super compact core.

In connection with the utilisation of neutron beam equipment, performance figures were calculated for the present core configuration at 2 MW power and for two upgrade options: A partial upgrade, keeping the present core configuration at 2 MW power, but extension with a cold neutron source and the beam tubes embedded in a beryllium block, and a full upgrade to a super compact core with 9 fuel assemblies, beryllium reflected at all sides, CNS and increased power level of 3 MW. A common figure-of-merit for the beam performance is the flux per unit wavelength at a wavelength of 0.45 nm (4.5 ?). With a value of 6.8x10^8 n/cm^2/s/nm at the neutron guide system exit for the fully upgraded HOR, it improves the present performance by a factor of 40. For the partial upgrade this factor amounts to about 14. The integral cold neutron flux at the neutron guide system exit amounts to 4.4x10^8 n/cm^2/s for the fully upgraded system, and 1.6x10^8 n/cm^2/s for the partial upgrade. From the results mentioned above, it can be concluded that a full upgrade of the HOR is feasible. It can be considered to be a viable option to increase the performance to the desired level. Also, depending on cost-benefit criteria, a partial upgrade may remain an attractive proposition. Compared with the upgraded reactor, the performance of the accelerator system remains at the level of the present HOR.

6. References

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