PERIODIC SAFETY REVIEW OF THE BR2 REACTOR

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Abstract

BR2 is a 100 MW tank type material test reactor which reached its first criticality on 29 June 1961 and is in power operation since January 1963. Since then, the reactor has been subject to an important number of maintenance and modernization projects. Its beryllium core has been replaced two times and a major power upgrade was made in 1972 by replacement of the primary heat exchangers. The operating license of BR2 prescribes a periodical safety review every ten year. According to the directives of the Belgian regulator (Federal Agency for Nuclear Control - FANC), the operator starts the periodical safety review by proposing a list of subjects to be evaluated. The subjects have to be long term items, which are not covered by the normal routine operation. These items could be the consequence of new practices, return of experience, important modifications, upgrades, ageing management … This article describes how the list of items are defined in cooperation with the authorities. The final list consisted of 19 items. For each of the selected subjects a detailed description of the issue was written. Each chart contained information about the safety relevance, the applicable rules and the shortcomings together with a plan for further action. The actions could be further analysis, change in organization or hardware modifications. The execution of the action plans is closely followed by the authorities. The two most important items are the follow up of the beryllium matrix and the inspection of the reactor vessel, since these two items are mentioned in the license. Some of the items which were analyzed led to improvements. Other items were subject of further analysis, but without significant modifications to the installation and a few items had to answer organizational aspects. A special subject is the conversion of the core to the use of low enriched uranium. Due to events in Fukushima, Japan, and the questions about the resistance of nuclear reactor against severe external events, a last paragraph will be devoted to the application at BR2.

1. INTRODUCTION

The operating license of the nuclear installations of SCK•CEN has no expiration date. However, it has a condition that the operator must submit every ten year a report about the safety of the installations, including the nuclear reactors. This condition could be considered as the implementation of safety guide NS-R-4, Safety of Research Reactor, article 2.2 about the safety objectives. According to the directives of the Belgian regulator, Federal Agency for Nuclear Control (FANC), the operator starts the periodical safety review by proposing a list of subjects to be evaluated [1]. The subjects have to be long term items, which are not covered by the normal routine operation or maintenance. These items could be the consequence of new practices, return of experience, important modifications, upgrades, ageing management.

2. DESCRIPTION OF BR2

The BR2 reactor is a heterogeneous thermal high flux engineering test reactor, designed in 1957 for SCK•CEN by NDA [Nuclear Development Corporation of America, White Plains (NY, USA)]. It has been built on the site of the SCK•CEN laboratories in Mol, Belgium. First criticality of the reactor was obtained in June 1961 and routine operation started in January 1963.

The reactor is cooled and moderated by pressurised light water in a compact core positioned in and reflected by a beryllium matrix. Up to now the reactor is operated with high enriched uranium. A conversion to low enrichment is planned by 2016 on the condition that fuel plates with sufficient uranium density will be qualified and available. The maximum thermal flux approaches 10^{15} $cm^{-2}s^{-1}$. During the lifetime of BR2, there have been a number of inspections, refurbishments and modifications [2]. The major modification is the increase of the ultimate cooling capacity in 1971, from originally 50 MW up to 125 MW. The reactor is generally operated at a power level between 50 and 70 MW.

The general lay out is given in *Figure 1*, which gives a cross cut view of the containment building and the machine hall. In the containment building, the reactor vessel is shown as located in the reactor pool. The floors for experimental devices are located around the reactor pool. The room under the lower vessel head is accessible and allows the introduction of experimental loops. The figure gives further a section of the hydraulic channel for spent fuel elements and a section of the hot cell. Both are located in the machine hall. The containment building can be isolated in case of release of radioactivity. However a major part of the primary circuit, namely the primary heat exchangers, the pressurizer, the primary pumps and the primary purification circuit are located outside the containment building. Automatic valves are foreseen in the primary circuit which can isolate the part of the circuit in the containment building from the outside in case of incident with potential radioactive release.

2.1. BR2 main data

- Beginning of utilization: 1963;
- Maximum heat flux:
  - Routine operation: 470 W/cm²;
  - Maximum admissible: 600 W/cm²;
- Nominal power: 60 to 100 MW;
- Maximum neutron flux (for 600 W/cm²):
  - Thermal: 1.2×10^{15} $cm^{-2}s^{-1}$);
  - Fast (E > 0.1 MeV): 8.4×10^{14} $cm^{-2}s^{-1}$;
- Irradiation positions: up to 100;
- Fissile charge at start of cycle: 10 to 13 kg ^{235}U;
- Operation cycle:
  - Minimum 7 days shut-down;
  - Nominal 21 or 28 days operation;
  - Possibility of short cycles;
- Days full-power operation per year variable, presently between 105 and 140 days/year.

BR2 is at this moment for the main part utilized for:

- Production of ^{99}Mo by the irradiation of uranium targets, for which 6 irradiation baskets are available;
- Production of isotopes for medical and industrial purposes by neutron irradiation;
- Irradiation of materials, both for nuclear power plants and for fusion projects;
- Irradiation of silicon crystals for semiconductor fabrication with two production devices: one tube loaded in the reflector for 5 inch and another one the pool near the vessel wall for 6 and 8 inch blocks;

— Testing of new fuel for research and test reactor in the framework of conversions to low enrichment;
— Testing of new power plant fuels, eventually in transient conditions.

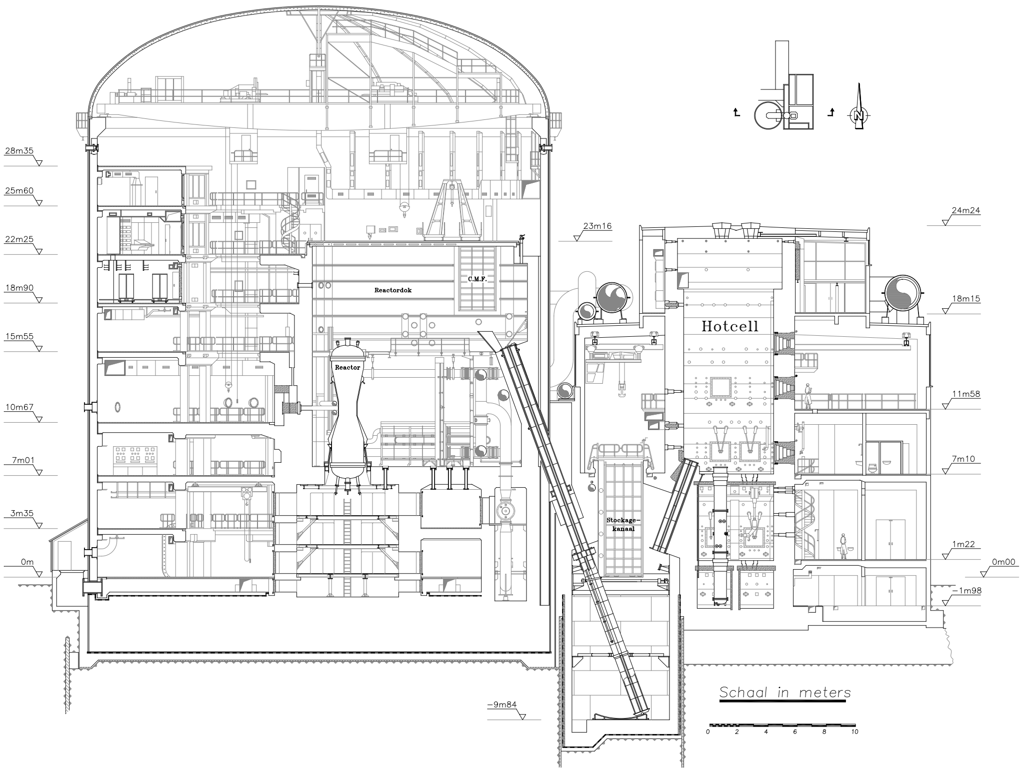

Figure 1. General lay out of BR2.

3. THE PERIODICAL SAFETY REVIEW

3.1. Choice of the subjects

For the review of 2006 an inventory (more than 70 items) of potential subjects was proposed by the operator. An internal selection of these items was made by the department of health and safety according to the criteria for the periodical safety review. For each of the selected subjects a detailed description of the issue was written. Each chart contained information about the safety relevance, the applicable rules and the shortcomings together with a plan for further action. The actions could be further analysis, change in organization or hardware modifications. The charts were transmitted to the Technical Support Organization of the FANC (BEL V) for further discussion. The final list contained 19 items. After approval by BEL V the charts are sent to the FANC. The execution of the actions defined in the charts is followed by BEL V during there regular inspection meetings and by the FANC during the half year meeting. Items with a generic interest for research reactors are described in this article.

3.2. Aging

3.2.1. The beryllium matrix

The beryllium matrix is a structural element of the BR2 reactor core. During irradiation, helium and tritium are formed in the beryllium. This has two adverse effects: swelling and neutron poisoning. The swelling is the cause for mechanical damage, while the neutron poisoning makes operation of the reactor more difficult. Both effects limit the useful life of the beryllium matrix.

The BR2 licence sets a limit on the lifetime of the beryllium. The first condition is that the matrix must be replaced if the inspections indicate that there is a risk for loss of material which could block the cooling of a fuel element or block the movement a control rod. The second condition gives an ultimate lifetime. This is reached when the fluence of fast neutrons (energy > 1 MeV) reaches 6.4×10^{22} cm^{-2} in the most irradiated channel. This value is derived from experience with the first beryllium matrix, which was irradiated up to 7.95 10^{22} neutrons/cm². This value was estimated as too high. Figure 2 gives an illustration of damage near the end of life of the first matrix. It was recognized that above 6.4 10^{22} neutrons/cm², the swelling goes faster than linear and the damage becomes too serious. Also an increased concentration of tritium in primary water was observed.

An inspection scheme is defined to follow up the matrix. At periodical intervals visual inspections and dimensional measurements are foreseen. Digital videos are made of the channels and the observed crack lengths are documented in order to compare with the results of previous inspections.

As a topic for the PSR a comparison was made between the second and the third matrix; the evolution up to now is comparable. First cracks are beginning to appear. The expected lifetime is the first half of the years 2020, depending on the utilization of the reactor.

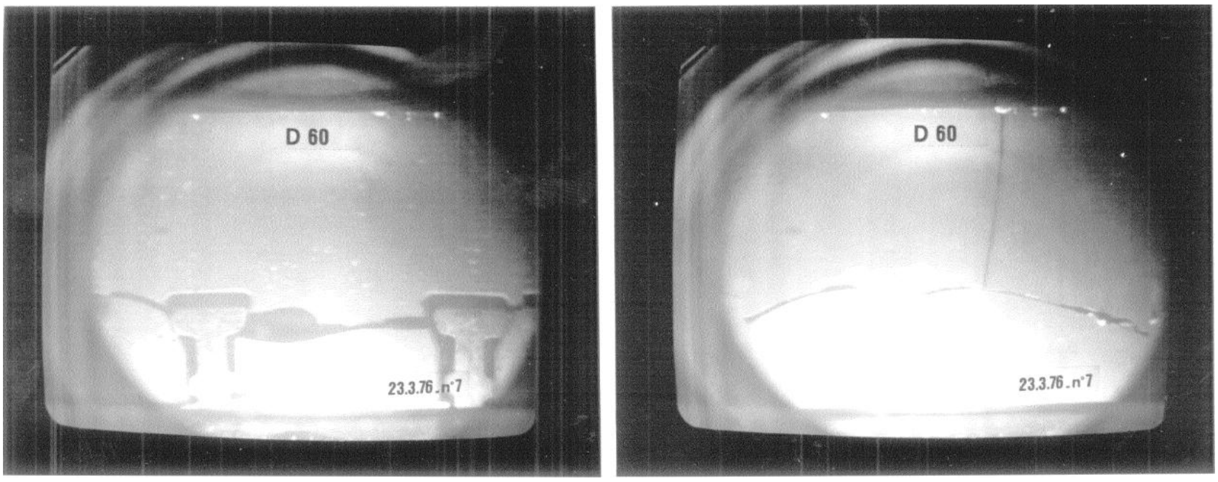

Figure 2. Damaged beryllium with material loss of the first matrix.

3.2.1. The reactor vessel

In the mid-nineties, on the occasion of the second matrix replacement, the vessel was inspected and a formal reactor vessel follow-up program was defined. The irradiated part of the vessel was completely inspected for the presence of cracks. For the non irradiated parts, the welds were inspected. This inspection was complemented by a fracture mechanical calculation of the stresses in the vessel taking into account the low cycle fatigue load of the start and stop of the reactor and a number of anticipated transients such as stop of the reactor with a fast pressure drop. In order to obtain the mechanical properties of the irradiated aluminium, samples were taken from the shroud mantle around the vessel. This has nearly the same irradiation history as the vessel material itself. Part of the samples were immediately tested. The rest are loaded in irradiation baskets in order to obtain samples with a higher dosed and are tested on regular intervals. The status of this program was evaluated in the framework of the PSR. The material evolves as predicted, such that the safety of the vessel is guaranteed for the next operational period (until 2016). Around this time, the replacement of the beryllium core is envisaged. On this occasion a non destructive inspection of the vessel wall will be done.

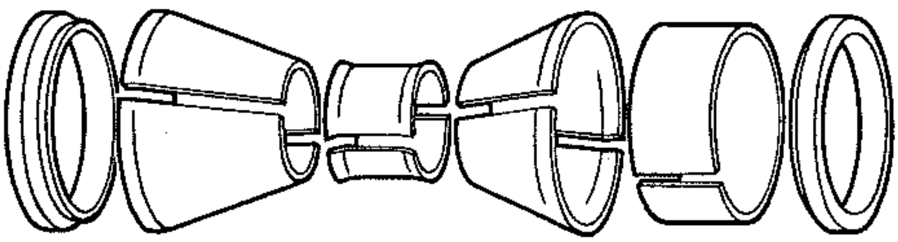

Figure 3. Components of the BR2 reactor vessel.

3.2.3. Control rods

The BR2 reactor core was designed with cadmium control rods. Around 2006, the cadmium was burnt too much and the absorbing parts had to be replaced. The absorbing parts were made of tubes of cadmium with aluminium cladding made by coextrusion of both materials. The original production method was no longer available and the decision was taken to use hafnium as an absorbing material. The advantage of hafnium is that it can be used in water, such that no cladding is necessary and the production is much easier. Another advantage of hafnium is that the main absorbing isotope is transmuted in another absorbing isotope. In this way, hafnium keeps is absorbing characteristics for a very long period [3]. Within the same project, all drive mechanisms and position indicators were replaced.

3.2.4. Additional inspection

In order to improve the follow up op periodical inspection a complete inventory of safety related inspection was made with the definition of the inspection period and the allowed delays.

Four additional special inspection programs were defined during the PSR:

— A special inspection program, based on the ASME XI code, is defined for the non irradiation part of the primary circuit. The program consists of non destructive inspection of the welds and the supports. The program will be executed during the next years;
— A number of devices, which were originally foreseen as experiments, became in fact permanent installations. Examples are the production baskets for molybdenum and the irradiation tube for silicon. These devices must be inspected as parts of the reactor;
— The high pressure irradiation loop for testing fuel and materials in PWR conditions was inspected and its safety documentation reviewed;
— A leak detection and monitoring system for the beam ports (no longer in use) was installed and mitigating measures for leaks are defined.

3.2.5. Replacement of components

A part of the radiation control equipment, especially the chain for measurement of the activity of the primary water, became unreliable and difficult to maintain. A replacement program for these chains is going on.

The same remark can be made about non nuclear instrumentation (measurements of flow, pressure, temperature). A number of this instrumentation gave also interruptions when the electrical power changed automatically from the normal feed to the diesel generators in case of disturbance on the external power grid. Improvements are being designed.

3.2.6. Probabilistic safety assessment

In the nineties, a probalistic safety assessment model of BR2 was composed [5], as a support for the refurbishment of the installation in 1996 during the second beryllium replacement. Later the model, which is based on event trees completed with fault three for the different systems that have to interact, has been completed. Various support systems were included. At this moment the PSA model is as complete as is reasonable possible. A subject of the actual safety review was to draw the final conclusions of the model.

3.2.7. Competence management

As most of this type of research reactors, BR2 is more than 50 years in operation. All persons involved in the design, construction, commissioning and initial operation have been gone. A threefold action was defined:

- All original plant documentation such as drawings, calculation notes and description was collected and archived in a systematic way. Nearly 90% of all the referenced information was found. For the remaining no dedicated search effort shall be continued. This action was necessary in order to have correct information of the original design base and the modifications during the operational life;
- A second action was defining formal training programs for operators and other personnel. In this way new recruited persons can do their tasks in an efficient way. For the other retraining programs are defined. Training can be given in 3 ways: theoretical lessons, practical training or on-the-job training;
- The third action is a way to learn from experience. An important component is learning from incidents that occurred in the own or in foreign, comparable installations. Other issues are documenting operating procedure with background information and making reports of unique but complicated tasks.

3.3. Reglementation and norms

Norms and regulations change during time and installation must be checked periodically against these new requirements. For the actual PSR, two items were concerned, namely the internal explosion risk and the hoisting cranes.

Since a few years, due to an European directive [4], it is required to check installation for the risks on internal explosions. The risk is found to be low. Except for a number of indications about potential explosive atmosphere, the installation was found to be conforming.

The hoisting devices in the reactor building all date back to the original construction. The crane in the adjacent machine hall had been upgraded in the nineties in order to take a loaded TN-MTR container with a weight of about 25 tonnes. However, none of these cranes fulfilled the single failure criterion as defined the NUREG guides [5]. A project was started to upgrade the crane. The upgrade of the polar crane (5 tonne and 20 tonne) is under way at the moment.

3.4. Conversion to low enriched uranium

A special subject is the conversion of the core to the use of low enriched uranium. In the framework of the PSR, the question had to be answered if it is possible and appropriate to convert BR2, and in case of a positive answer give the requirements for the fuel. The first analysis shows that using a fuel density around 8 gram per cc it is possible to operate BR2 without major loss of performance. The density can be reached using UMo dispersed fuel with and increased quantity of silicon. Test plates have been irradiated in 2010. The analysis is going on this moment. The test irradiations are performed as part of the European initiative on the qualification of high density low enriched fuel for research reactors [7]. However, since the program will extend beyond the time frame of the PSR, the subject is, after the first initial feasibility study, no longer considered as a PSR project. From the licencing point of view, the conversion is to consider it as a modification of the reactor. The licencing procedure and the content of the information are under discussion with the authorities

4. PROTECTION AGAINST SEVERE EXTERNAL EVENTS

After the events at the Fukushima nuclear power plant, various organizations requested for so-called stress tests. These are evaluations of the installation under severe external conditions such as beyond design earthquakes, extreme flooding and weather conditions [8]. The Belgian authorities decided, on request of the parliament, to include man made events such as air plane crashes, bomb attacks and cyber attacks. For the Belgian research reactors a progress report is to be made by mid December 2011 and the final report must be ready June 2012. In contrast with the power reactors, no peer review of the report is foreseen. The conclusion of the analysis could eventually lead to additional protection measures to protect the installations or to mitigate the consequences of potential damage.

5. FINAL REMARKS — CONCLUSIONS

The PSR project and its realization, which extend over a number of years, indicate that it possible to upgrade an old installation towards modern standards, although this requires a major effort and investment. However, it is not only the technical conditions of the reactor which determines its safety level. It is also important to keep the knowledge up to date, both on the side of the documentation of the installation and the knowledge of the people. This is especially important for reactors older than 40 years because people who have designed and constructed the installation will no longer be available.

Beside these issues which are due to ageing of the installation, a number of items are due to external factors. The conversion to the use of low enriched uranium is caused by the fact that high enriched uranium is no longer available. Even external events, such as the accident with the Fukushima nuclear reactors, can introduce subject for analysis which can lead to modifications.

REFERENCES

[1] BELGIAN FEDERAL AGENCY FOR NUCLEAR CONTROL (FANC/AFCN), http://www.fanc.fgov.be/nl/page/inleiding/130.aspx (text in Dutch or French).

[2] KOONEN, E., JOPPEN, F., GUBEL, P., “Safety challenges encountered during the operating life of the almost 40 year old research reactor BR2,” International Conference on Topical Issues in Nuclear Safety, Vienna, 2001, IAEA, Vienna.

[3] GOSSET, D., HERTER, P., (rédigé par M. COLIN), Matériaux absorbantes neutronique pour le pilotage du réacteur, Rapport CEA BN 3720.

[4] European Directive 1999/92/EG of December 16, 1999 (also known as ATEX 137).

[5] NUREG-0554: Single-Failure-Proof Cranes for Nuclear Power Plants (1979) and NUREG-0612: Control of Heavy Loads at Nuclear Power Plants (1980).

[6] JOPPEN, F., “Experience with Probabilistic Safety Assessment for the BR2 Reactor,” International Conference on Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management, Santiago, 2003, IAEA, Vienna (2004).

[7] FRERY, F., et al., “LEONIDAS UMo Dispersion Fuel Qualification Program: Progress and

Prospects," 32nd RERTR Meeting, Lisbon, 2010.
[8] See www.wenra.org, "Stress test" specifications – Proposal by the WENRA task force, 21 April 2011.