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SPECIFICITY IN THE LICENSING PROCESS OF REDUCED ENRICHMENT IN THE BULGARIAN RESEARCH REACTOR

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Presented at the

RERTR 2004 – International Meeting on Reduced Enrichment
for Research and Test Rectors

7 – 12 November 2004
International Atomic Energy Agency
Vienna, Austria

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ABSTRACT

The presented paper considers some specific questions of the licensing process regarding the reconstruction of the Bulgarian IRT-2000 research reactor, which includes conversion to the low enriched fuel. This specificity has risen as a result of two facts. The design of the reactor reconstruction was made on the basis of the existing fresh 36% highly enriched fuel. But after finishing of the design process, this fresh highly enriched fuel was shipped back to Russia in the framework of the RERTR program. These facts have involved some changes in both – in the licensing and the design processes. Re-analysis of the neutronic and thermal-hydraulic calculations is required to be made on the base of the technical specifications of the new LEU fuel. To facilitate the licensing process the NRA has adopted regulatory acceptance criteria for approval of the reactor core design with LEU fuel.

1. Introduction

The research reactors core conversion to low enriched (LEU) fuel and utilization of a newly developed fuel will be the essential challenge to the licensing and operation of these reactors. Some risk of suspending the operating license of the older research reactors can be provoked by their inability to meet new safety requirements. Many research reactors, which are under extended shut down conditions, also can be converted to the low enriched fuel but the conversion process should be adapted to the current state of the reactor installation. To keep lower costs and to satisfy the safety requirements it is reasonable to accomplish reducing of the fuel enrichment in parallel with refurbishment and modernization of the old research reactors undertaking special measures for shipment of the cumulated spent fuel and for radioactive waste management. This leads to some specificity in the licensing process of reduced enrichment for such reactors.

This paper presents the Bulgarian regulatory approach regarding the IRT-2000 research reactor refurbishment giving a special attention to the conversion of the reactor core to the 19,7 % LEU fuel – IRT-4M.

2. Licensing status of the Bulgarian research reactor

The IRT-2000 research reactor (IRT RR) was designed and built from 1959 to 1961 as a pool type reactor. First criticality was reached in September 1961 and it was put into normal operation in November 1961 according the Decree of Council of Ministers. At that time the reactor was in operation without license for operation due to a lack of a Regulatory Basis.

The reactor was shut down on 13 July 1989 according to prescript of the Regulatory Body (at present – Nuclear Regulatory Agency). In this prescript many remarks and additional requirements had to be responded in order to enhance the safety in reactor operation. Although the operator – Institute for Nuclear Research and Nuclear Energy – met the requirements put into the prescript, the reactor remained in extended shut down state during the last 12 years. The main reason for this extended shut down was absence of any Government decision for the future of the reactor till 1999. On 17 May 1999 the Council of Ministers decided to cease the operation of the IRT-2000 reactor until a final decision is taken on the basis of site investigation for further use of the reactor installation and the IRT-2M fresh fuel. The Decree of July 6, 2001 of the Council of Ministers enacts to reconstruct the old reactor into a 200 kW (low power) reactor.

On 25 September 2002 the operator of the IRT RR applied for permit for design of a low power (200 kW) reactor. The NRA has issued a permit to INRNE, on 17 December 2002, to design a low power reactor, which includes dismantling of the existing IRT-2000 reactor systems. On 22 December 2003 the operator has submitted to the NRA a Preliminary Safety Analysis Report (PSAR) and on 25 April 2004 – a design project for the low power reactor. The design project was based on the 36% HEU fresh fuel IRT-2M fuel - Russian origin, which was received in Bulgaria in 1980. But in fulfillment of the program for shipment of the HEU fuel, this fresh fuel was shipped back to the Russian Federation at the end of 2003. This fact provoked confusion into the licensing process.

3. Specificity in the licensing process

The new Bulgarian Act on the Safe Use of Nuclear Energy /ASUNE/ was adopted in the year 2002. It replaced the former Act on the Use of Nuclear Energy for Peaceful Purposes, which was in force for about 17 years.

The ASUNE covers the activities involving nuclear energy and sources of ionizing radiation mainly by establishing a consistent authorization regime. It is an up-to-date act, based on the IAEA requirements and standards, and yet fully in compliance with the Bulgarian legislative system.

The newly adopted Act is based on the following main principles:

- Priority of safety over economic and other social needs;
- Occupational and public exposure to ionizing radiation to be kept as low as reasonably achievable /ALARA/;
- Direct and personal liability of the licensee/permit holder;
- Independence of the regulatory body;
- Application of a less prescriptive approach;
- Issuing of authorizations under conditions of legal equality and transparency.

The ASUNE prescribes issuing of two types of authorizations:

- Licenses;
- Permits;

From a legal point of view those two types of authorizations have one and the same nature – they are individual administrative acts according to the Bulgarian law. That's why there is no difference between those authorizations in terms of the issuing procedure.

Licenses and permits are issued following a submission of application with enclosed documentation. The main questions concerning nuclear safety and radiation protection, as well as safety analysis, are considered in the licensing documentation with the aim to reduce the risk from improper and unauthorized use. Fuel and core design features are included in the Safety Analysis Report (SAR). The SAR includes information about type, amount, physical, thermal and hydraulic characteristics of the fuel system, fuel load patterns and the corresponding operational limits.

The NRA staff should conduct analyses and safety assessments on the basis of information included in the SAR to evaluate the adequacy of design and safety features of reactor systems.

To manage this process, the NRA has established an internal procedure, which is a document from the NRA Quality Assurance Program. The flow chart diagram of this procedure is shown in Fig. 1. According to this procedure the applicant, who has applied for a license or permit, is obliged to submit to the Chairman a complete set of the licensing documentation, which is defined in a separate regulation. The Chairman distributes this set of documents to the General Department on Regulation of Safety in Nuclear Facilities and the Department on Safety Analyses, Assessment and Research. With the aim to conduct a safety assessment of the enclosed documentation, the heads of these two departments compose a Safety Assessment (SA) team. The SA team evaluates the information, included in the licensing documentation, in respect to:

a) Completeness – All significant influence on safety must be identified and adequate safety measures should be included in the licensing documentation. Any additional risk foreseen but not specifically analyzed or protected against must be shown to be negligible.

b) Clarity – There must be a presentation of the processes and the safety justification that will be applied, with clear reference to supporting information and clear identification of conclusions and recommendations.

c) Objectivity – The conclusions in the safety assessment should be supported as far as reasonably practicable with factual evidence. The understanding of the systems behavior or processes should be established from appropriate research and development.

d) Correctness – The methods and codes, that are used to demonstrate safety, must be developed for this purpose.

Concerning the fuel utilization and core arrangements, the SA team evaluates the information included in the licensing documentation to confirm that the functional capabilities of the fuel system are not reduced below those assumed in the safety analysis.

To manage the IRT refurbishment licensing process has applied the procedure described above but taking into account the specificity of the current status of the reactor. The specificity has risen as a result of the following three facts:

- The operator does not possess an actual license for operation;
- The reactor core was designed on the basis of 36% HEU fuel – IRT-2M;
- This fresh 36% HEU fuel was shipped to its country of origin in compliance with the requirements of the RERTR program after completion of the design process.

To avoid these problems and to facilitate the licensing process the NRA applied a little bit different procedure, which flow chart is shown on Fig. 1.

The NRA decided to use an independent assessor to support the licensing process. The independent assessor evaluates safety analyses performed by the manufacturer and operator in respect to the fuel design criteria applying their knowledge and performing independent calculations. The use of different computer codes from the assessor and operator is needed to perform a real independent assessment. The accomplishing of an independent expertise is a separate step in the licensing process. The independent assessor should prepare a document with expert conclusions and recommendations. On the basis of such document the regulatory authority should issue an order for approval of the IRT-200 design, including utilization of the new LEU fuel design [1]. The NRA also required from the operator to apply similar approach and to employ an independent assessor. This assessor should be involved in the design process and to do ad-hoc assessment of the design during the design process. The operator is obliged to implement all reasonable comments and recommendations of the assessor. From the NRA side such independent assessor is the UK Serco Assurance Ltd. and from the operator side – Belgatom, Belgium.

The information provided in the PSAR and associated documents are the primary basis for licensing and the NRA should make a determination regarding the following points:

- Provision of sufficient and adequate information;
- Compliance of information with all regulatory rules and regulations;
- Accuracy of information (i.e. independent checks of design and quality assurance programs);
- Feasibility and capability of engineered solutions to meet the design objectives with regard to nuclear safety and radiation protection.

To asses the information included in the SAR, the NRA has to apply acceptance criteria. For that purpose the NRA has developed criteria for acceptance of the SAR [2]. These criteria are divided on:

1) Criteria for completeness of the PSAR

- Completeness of the PSAR contents and structure;
- Completeness of the postulated initiating events list;

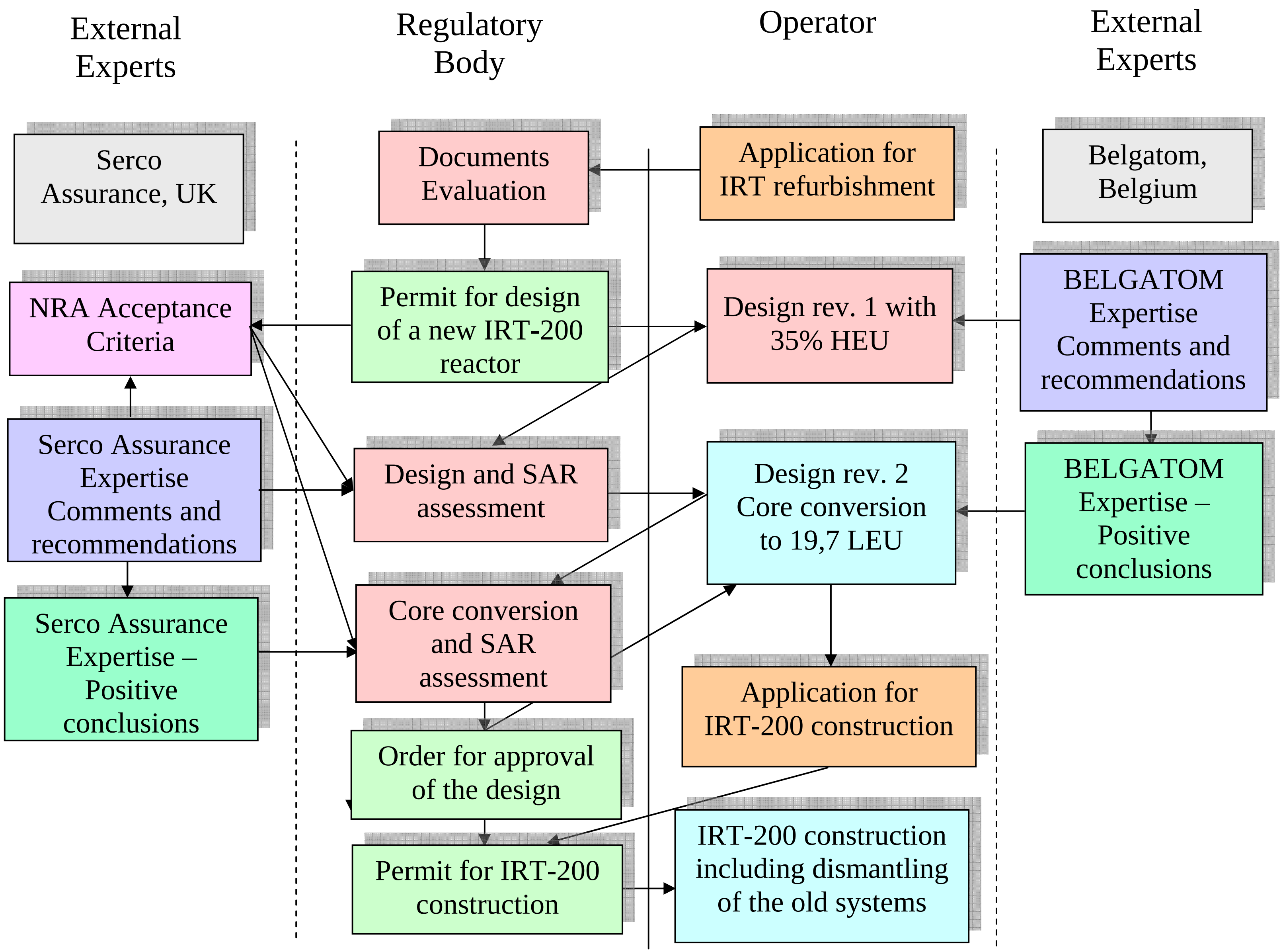

Fig. 1: Flow chart of the licensing process for refurbishment of the IRT-2000 RR

2) Criteria to the information included in the SAR

- General description of the facility
- Design bases, including nuclear fuel and reactor core design;
- Design information on all components of the reactor core;
- Design information on reactor systems and components;
- Systems or component description, including drawings, schematics and specifications of principal components, including materials;
- Operational analyses and safety considerations;
- Instrumentation and control features;
- Technical specifications requirements and their bases including testing and surveillance.

3) Radiological criteria

- Radiological criteria for normal operation – ALARA levels;
- Radiological criteria for accident conditions – dose limits for operator and general public and gaseous and aerosols release limits to the environment.

According to the national legislation, the NRA conducted licensing activities to review the IRT RR licensing application and evaluated the proposed SAR, which was based on 36% HEU fuel. To determine whether or not the facility can be reconstructed and operated consistent with the applicable regulations, the NRA used these acceptance criteria during the evaluation process. On the basis of design assessment, analysis of fulfillment of defined acceptance criteria and results from the external expertise, the NRA returned the design project to the operator for new revision. It should be noted that the completeness of the regulatory review was itself hampered by the deficiencies in the SAR, so that a further review of the revised Safety Case should be carried out once these deficiencies have been addressed.

This new revision should include the core conversion from 36% HEU to 19,7 LEU fuel and should consider all comments and requirements made by the independent assessors. Before starting of the IRT-2000 refurbishment the operator should fulfill the following regulatory requirements:

- Revising of the design project and the safety analysis report for removal of all comments and re-analysis of core with respect to LEU core conversion as well as re-analysis of the safety case;
- Obtaining new external expertise from both the operator and the regulator assessors with positive conclusions for adequacy of the design on the basis of LEU fuel;
- Shipment of the cumulated HEU and LEU spent fuel back to the Russian Federation.

4. Regulatory requirements to the revised design project

Summary of reactor facility changes

For clear understanding of the design changes it is preferable to include a new chapter in the SAR in which the operator will provide a brief summary of any proposed changes with references for the detailed fuel design and performance discussions. Additional useful information could be provided by a comparison between the IRT-3M HEU, on which the first revision of the design was based, and the new IRT-4M LEU, which will be used in the converted reactor core.

Also any changes in the safety analyses, safety margins and operational limits and conditions should be briefly listed with a reference to the SAR sections in which they are discussed in more details.

Comparison with similar facilities already converted to LEU fuel

A comparison with similar facilities that have been already converted to LEU fuel will give additional evidences for successfulness of the conversion process. The information should be oriented to the similarities and differences between the facilities in design, and construction. Any problems that were identified and resolved at the converted facilities can be briefly discussed and measures to avoid such problems at the IRT-Sofia facility should be addressed in that part of the SAR.

Reactor core design

A detailed discussion of the reactor core components and structures should be provided by the operator, including a summary description of the core changes for the conversion to LEU fuel. The figures and lists comparing important design parameters and operational characteristics should be also provided. The SAR should outline principles for selecting allowable core configurations.

The LEU fuel elements should be compared to the HEU fuel elements. Any changes resulting from the lower enrichment and possible higher uranium concentration in the LEU fuel elements is required to be included in the SAR. The operator should discuss in details the mechanical design of the fuel elements, volume ratios of fuel to moderator and fuel to coolant, the thermal capabilities and characteristics of fuel components.

The anticipated inherent reactivity feedback coefficients, including the reactivity coefficients of the fuel temperature, moderator temperature, moderator density and voids, and the power distribution variations should be provided. The characteristics and mechanisms of a prompt fuel temperature coefficient of reactivity and its effect on stability and safety of the reactor operation should be analyzed. Changes in delayed neutron fraction and prompt neutron lifetime resulting from the core conversion should be also considered. The analyses should also include the plutonium production and effects of Plutonium-239 on LEU reactor operating characteristics, such as changes in reactivity and delayed neutron fraction.

Control rods worth and excess reactivity

Any changes to the characteristics of the control rods (e.g. the mechanical design, material, and configuration) for the proposed LEU core should be described and excess reactivity should be analyzed. The safety analysis should ensure that control rod worth for the proposed LEU core remains within acceptance limits for control or shutdown functions or for potential reactivity change accidents.

The factors as the effects of Xenon-135, Samarium-149, and other fission products, void coefficient, temperature coefficients of reactivity for fuel and moderator, burnup and generation of fissile material impact on reactivity from the experiments should be considered in the revised version of the design. The neutron spectral hardening associated with conversion to LEU should also be considered.

Changes should be analyzed to demonstrate that reactor control and function should be acceptably ensured. The control rod worth and excess reactivity should be verified in the commissioning tests.

Shutdown margin

The shutdown margin is an important parameter because it relates to the capability of the control system to shutdown the reactor safely under any operating conditions. The operator should present the bases for shutdown margin technical specifications for the LEU core and compare them with those stated for the HEU core. Any changes in the shutdown reactivity for the cold, clean reference core arrangements and operating conditions that may be limiting with regard to shutdown margin should be analyzed in the SAR. The safe shutdown of the reactor should be demonstrated, including any reactivity changes caused by expected burnup and by failure of any allowed movable experiment.

Thermal-hydraulic characteristics

Any changes in the thermal-hydraulic parameters and characteristics during routine and transient conditions that may arise from the fuel conversion should be analyzed and presented in the revised version of the SAR. Possible changes include the number or dimensions of coolant channels and fuel tubes, core dimensions, power density, fuel and cladding temperatures, surface heat flux, radiation heating, thermal conductivity of the fuel and coolant flow rate. Changes to power peaking factors for the conversion to the LEU core should be included for the expected and limiting control rod operating conditions in this analysis. The results from the analyses should show that the LEU-fuels reactor thermal-hydraulic design will function under postulated accident scenarios and conditions. Any change in the thermal-hydraulic parameters between the HEU core and the proposed LEU core should be discussed in the revised SAR. Changes in the thermal-hydraulic characteristics should not result in exceeding design limits, such as departure from nucleate boiling ration (DNBR), flow instability, or fuel safety limits. If there would be significant changes in fuel element flow rates or heat transfer characteristics, verification should be provided that the changes result from factors in the conversion process beyond the control of the operator.

Any proposed changes to the technical specifications, including safety limits, should ensure that the reactor will operate safely during routine and transient conditions under all analyzed combinations of system parameters.

Accident analysis

The operator should present the comparison of the HEU and LEU accident analyses. This comparison should demonstrate that the conclusions reached in the safety analyses for HEU core remain valid for the LEU core. For example, a maximum beyond design basis accident that assumes a fission product release scenario for a HEU fuel element may be used for the LEU fuel if the operator demonstrates that the two fuels have sufficiently similar physical characteristics and fission product inventories.

New or revised accident analyses of the LEU fuel should be performed if the techniques used for the current HEU analyses are no longer available or appropriate. If the conversion creates the possibility of a new accident scenario or significantly changes the potential consequences of a previously accepted HEU accident scenario should be also considered in the SAR. New or revised safety analyses should demonstrate that the LEU core can safely withstand the postulated accidents and that the consequences meet the radiological acceptance criteria.

The revised version of the SAR should include a systematic approach for hazard identification. In the first revision of the SAR the hazard identification process was not carried out at a sufficiently detailed level to identify all possible fault sequences. Instead, a list of generic hazards was used as the basis of the safety assessment. In some cases, additional hazard identification has been performed to identify the individual initiating events, which could cause the 'generic' fault; this has not been performed systematically to ensure that the listing is complete and comprehensive. The main recommendation that is made by the Serco Assurance [3] is:

"To specify the classification of fault sequences and structure the analysis of these sequences to clarify the assumptions and operating envelope, and improve the rigor of the safety argument to demonstrate that hazards have acceptable consequences or very low frequency".

With regard to Design Basis and Beyond Design Basis Accident Analyses, there are a number of general comments that are made by the Serco Assurance [3]. The following are considered to be significant deficiencies in the treatment given in the SAR:

- Criteria are unclear for classification of faults as within or outside the Design Basis;
- There are insufficient analyses of initiating event frequencies and radiological consequences (i.e. doses to public and workers) for fault sequences to enable them to be classified;
- For Design Basis faults, there is insufficient analysis of event sequences;
- For Design Basis faults, there is insufficient justification that the safeguards provided are sufficient and that the design is tolerant of single failures;
- From the analysis of Design Basis faults, there is no clear definition of the key safety systems and the safe operating envelope (see further detailed comments);

- For Beyond Design Basis faults, there are inadequate Accident Analyses.

The operator should clarify the criteria for allocation of events as Beyond Design Basis and indicate clearly which accident sequences are considered in this category.

Alternatively, the accident analyses should demonstrate adequate assurance of public health and reactor safety for the proposed core and fuel. Also, the accident analyses should show that the radiological acceptance criteria are met or compensatory measures are proposed to ensure that the acceptance criteria will be met.

Requirements for computer codes

Validation evidence for all codes used in the SAR should be presented and reasons for using different codes in different parts of the safety analysis should be outlined. The verification status of used computer codes for physical characteristic and control rod efficiency calculations during the design and accident analysis has to be presented in the SAR [3].

5. Conclusions

Many activities for investigating the economic and safety implications in future reconstruction of the reactor are carried out. Many works for clarifying and subordinating the variety of activities have also been done in order to produce a reliable design project. As a result of these activities the design of the new IRT-200 low power reactor has submitted to the NRA for assessment and approval. But unforeseen circumstances as shipment of fresh HEU fuel, lack of design information concerning the new LEU fuel and lack of experience in the designing of research reactors led to some incompleteness of the safety justifications in the design. Many of the shortcomings found in the safety assessment derive from the lack of full identification of all fault sequences and initiating events. As a result of this the SAR does not demonstrate the safety of the design of the reconstructed IRT-200 reactor with sufficient rigour.

On the basis of mentioned above the NRA decided that the IRT-200 design should not be approved until the deficiencies in the SAR are remedied. In parallel with this the NRA proposed to the operator to revise the design project and SAR and to include in this new version the core conversion from HEU to LEU fuel.

References:

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