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ARTICLE

JRTR, the First Research Reactor in Jordan: Results of Commissioning in Light of Safety Enhancement Following Fukushima-Daiichi Accident

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Abstract: The Jordan Research and Training Reactor (JRTR) is a multipurpose, 5 MWth upgradable to 10 MWth reactor. Currently, the JRTR is in the operational phase. Prior to the start of JRTR operation, a set of commissioning tests have been performed. The IAEA safety guide NS-G-4.1 has been followed. The commissioning process was divided into three main stages with hold points at the end of each stage. These stages are; tests prior to fuel loading, fuel loading tests and initial criticality tests which include low power tests. The last stage constitutes power ascension tests and power tests up to rated full power. The performed tests proved that all design and performance parameters have been achieved. For instance, the thermal power of 5 MW, maximum thermal neutron flux of 1.5×10^{14} (n/cm^2.s) and negative reactivity feedback have been achieved. The safety of the JRTR was under extensive inspection from all involved parties. Particular attention has been paid to the lessons learned from the Fukushima-Daiichi accident and the recommendations made by the national regulator, the IAEA, the consultants and the owner. For instance, all safety aspects of the JRTR fall under the category of SC-3 according to the ANSI/ANS 51.1 classification system of nuclear reactors. As examples, the Reactor Structure Assembly (RSA), Primary Cooling System (PCS), Second Shutdown Drive Mechanism/ Control Rod Drive Mechanism (SSDM/CRDM), Reactor Protection System (RPS), Confinement Isolation Dampers, Siphon Breaking Valves and UPS have been classified as Safety Class (SC-3) components. Design changes of systems and equipment due to the reinforced international safety norm, addition, expansion and modification of facilities have been implemented. The quality class of several components, such as Process Instrumentation and Control System (PICS), Radiation Monitoring System (RMS), Information Processing System (IPS) and Operator Work Station (OWS), has been upgraded. Moreover, expansion and modification of facilities to accommodate systems and equipment have been applied. The seismic monitoring system has been improved by upgrading the quality class and by adding a function generating the automatic seismic trip signal when a seismic motion exceeds Operating Basis Earthquake (OBE). Pool liner integrity has been also enhanced. Furthermore, the emergency conditions have attracted special attention. The emergency water storage capacity has been increased and two mobile diesel generators have been placed in a building of seismic category I. This paper is divided into two main parts. The first part presents the commissioning stage of the JRTR as well as the final results and conclusions. The second part describes the safety aspects and the improvements made taking the lessons learned from the Fukushima-Daiichi accident into account.

Keywords: JRTR, Research reactor, Reactor commissioning, Reactor safety.

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Introduction

Similar to the goals of commissioning process for all research reactors, the objectives of commissioning of the JRTR are clear and definitive. These include: verifying that the SSCs are commensurate with their importance to safety, demonstrating that the design requirements are met as stated in the Safety Analysis Report [1], providing basic data for safe and reliable operation, verifying that documentation is adequate for full facility operation, providing operation staff with the chance of education for the validity of the reactor operation procedures and providing the end-users with a clear idea about the facility characteristics [2]. It is needless to say that one of the most important objectives of reactor commissioning is to verify the adequacy of facility operation under all anticipated operational modes. The commissioning of the JRTR is significant and of a panoramic importance, as several lessons have been learned from Fukushima-Daiichi accident. Commissioning is a true chance for testing the measures of safety that have been implemented as a result of the discussions between the involved parties. It is important to highlight that the reactor design, development, utilization, nuclear and radiation safety and nuclear security comply with the Jordanian laws and regulations at work. Additionally, the applicable standards and guidelines as set in the International Atomic Energy Agency (IAEA) Safety Requirements, NS-R-4 [3], US NRC report NUREG 1537 PART 1 [4] and Korean regulations and guidelines are used as a top tier requirement. In this context, the concept of defence in depth is applied in the design to provide protection against various reactor transients, including transients resulting from equipment failure and human error and from internal or external events that could lead to a Design Basis Accident (DBA). Particularly, the design of the JRTR satisfies the following criteria:

- The use of conservative design margins, the implementation of a quality assurance program and the organization of surveillance activities;
- The provision of successive physical barriers to the release of radioactive material from the reactor;
- Application of the single failure criterion by ensuring the fulfillment of each of the basic safety functions. The three basic safety functions are: shutting down the reactor, cooling, in particular the reactor core, and confining radioactive material. The essential characteristic functions associated with Systems, Structures and Components (SSCs) must ensure the safety of the reactor. In normal operation, the equipment needed to perform safety functions consist of the operating systems, which must be supplemented by other engineered safety features to perform their functions for anticipated operational occurrences and in DBAs;
- In the design of the safety systems, including engineered safety features that are used to achieve the three basic safety functions, the single failure criterion must be applied;
- Acceptance criteria are established for operational states and for DBAs. In particular, the DBAs considered in the design of the JRTR and selected Beyond Design Base Accidents (BDBAs) have been identified for establishing acceptance criteria. For the design of SSCs, acceptance criteria in the form of engineering design rules have been used;
- Shutting down the reactor and maintaining it in a safe shutdown state for all operational states or DBAs;
- Providing for adequate removal of heat after shutdown, in particular from the core, is included in DBAs;
- Confining radioactive material in order to prevent or mitigate its unplanned release to the environment;
- Inherent safety features, like the appropriate choice of materials and geometries to provide prompt negative coefficients of reactivity have been implemented during the design;
- The use of on-site and off-site emergency plans aimed at mitigating the consequences to the public and the environment in the event of a substantial release of radioactive effluents.

Commissioning of the JRTR

Commissioning Plan

Based on the guidelines of research reactor commissioning in Ref. [2], the commissioning plan of the JRTR has been envisaged to address the objectives of commissioning [5]. The plan defines the objectives of commissioning and the main chapters describe commissioning organization, stages, schedule, management, quality assurance, operational limits and conditions, radiation protection and emergency and security management during commissioning. For the purpose of conducting commissioning activities, the commissioning organization structure has been set. The structure clearly defines the commissioning groups, the functional responsibilities, levels of authority, approval channels and interfaces between the participating groups. Therefore, the organization structure of commissioning has been designed and implemented during the commissioning stage. The organization chart is presented in Fig. 1. It is composed mainly of the management group, commissioning group, reactor operation group, construction group, quality assurance group, safety and security group and safety committee. The functions and duties are clearly defined in the commissioning plan. For example, the management group, which is chaired by the JAEC Project Manager (PM) consists of KAERI PM who chairs the commissioning safety group, DAWEOO site PM, and JAEC reactor manager. The responsibility of this group is to provide strategic oversight and resources for commissioning, which includes: authorizing the official start of commissioning and declaring the acceptance of commissioning results, reviewing the commissioning plan and monitoring its implementation, following the NCRs and the appropriate corrective actions and coordinating between the commissioning groups. The group also plays a vital role in providing resources and making lines of communication between all relevant groups and parties. For details on the functions and responsibilities, the reader may refer to Ref. [5].

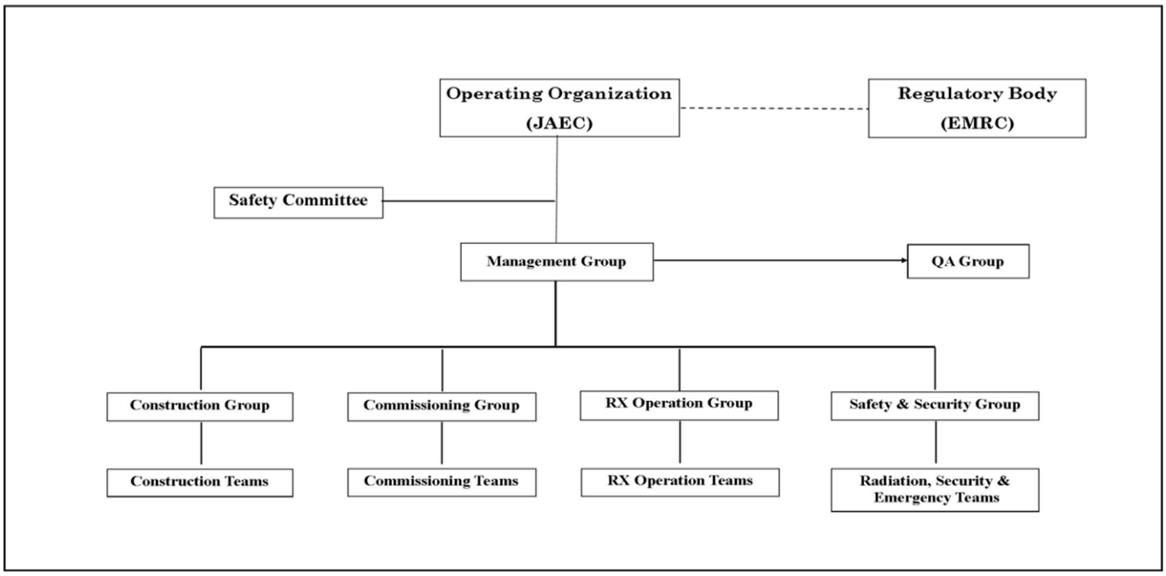

FIG. 1. JRTR commissioning organization structure.

Commissioning Experiments and Results

Following the commissioning plan described in [5], the commissioning activities have been divided into several stages. Preloading commissioning, Stage A, consists of three main stages: Stage A1, Construction Acceptance Tests (CATs), consists of tests distributed over the mechanical, electrical and I&C tests [6], while Stage A2, Flushing and System Performance Tests (SPTs), consists of flushing of the fluid systems and SPT for the systems reported in Reference [7] and finally Stage A3 [8], which consists of Integrated System Tests (ISTs). This latter stage A3 focuses on the simulation of the reactor operation during power and training modes. These two modes have been tested using simulated reactor power signals. A loss of power scenario also was simulated in this stage A3.

Table 1 presents the major planned hot commissioning experiments. Some of these experiments belong to the fuel loading and low-power tests (B1 and B2 stages). Other experiments are planned for the power ascension and full power tests (C1 and C2 stage). The initial JRTR core constitutes of 18 FAs with various uranium densities distributed around the core, as shown in Fig. 2. In this report, a summary of the main hot commissioning tests is presented.

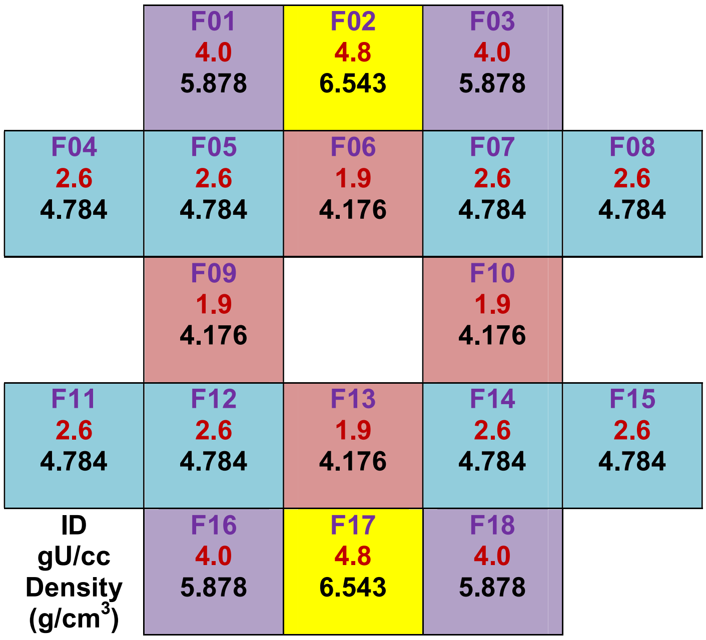

FIG. 2. Sketch diagram representing the JRTR core.
NOTE: Fuel assemblies with identification number and uranium density are illustrated.

TABLE 1. Planned tests for the hot commissioning phase.
NOTE: The stage in which each test is conducted is indicated.

Test	Stage
Fuel loading and approach to criticality	B1
Excess reactivity measurement	B1
CAR/SSR rod worth measurement	B2
Measurement of kinetic parameters	B2
Measurement of void reactivity coefficient	B2
Measurement of flux distribution	B2
Measurement of isothermal temperature reactivity coefficient	B2
Training mode operation	B2
Natural circulation test	C1
Neutron power calibration	C1
Measurement of power reactivity coefficient	C2
Measurement of xenon reactivity	C2
Shutdown and monitoring capability of the SCR	C2
Cooling performance test of PCS and HWS heat exchangers	C2
Cooling tower capacity test	C2
Thermal neutron flux at IR0	C2
NAAF performance test	C2
RI production test	C2
Loss of primary flow test	C2
Loss of normal electric power test	C2
Radiation surveys to determine shielding effectiveness	C1,C2
I&C function tests during operation	C2

Fuel Loading and Approach to Criticality

The test aims at reaching the initial critical core using the 1/M (inverse multiplication) method by insertion of external neutron source in the subcritical core and replacing aluminum dummy fuel assemblies in the core with actual fuel assemblies one by one. For details on the process, the reader can refer to Ref. [9]. The initial critical core is defined as the core having the minimum number of fuel assemblies necessary to achieve criticality. This initial critical core will be expanded to the first cycle operation core by loading additional fuel assemblies at the next test "excess measurement". The fuel assembly density and order of insertion for the initial core is presented in Table 2. The test also checks whether the initial criticality can be achieved at the initial critical core predicted by calculation.

The results of the test are shown in Fig. 3. In the figure, the count rate of the BF3 detector (counts per second) as a function of time (second) for the CAR position at 570.1 mm for the initial core of 14 fuel assemblies is presented. The reactivity ($) is also shown in the figure. The minimum critical core consists of 14 fuel assemblies as presented in Table 2 and the critical CAR position is 570.1 mm.

TABLE 2. Fuel assemblies including the uranium density (gm/cm^3) and order of loading for the initial critical core.

Fuel assembly	Uranium density (g/cm^3)	Order of insertion
F07	2.6	1
F12	2.6	2
F14	2.6	3
F05	2.6	4
F13	1.9	5
F06	1.9	6
F10	1.9	7
F09	1.9	8
F02	4.8	9
F17	4.8	10
F03	4	11
F16	4	12
F01	4	13
F18	4	14

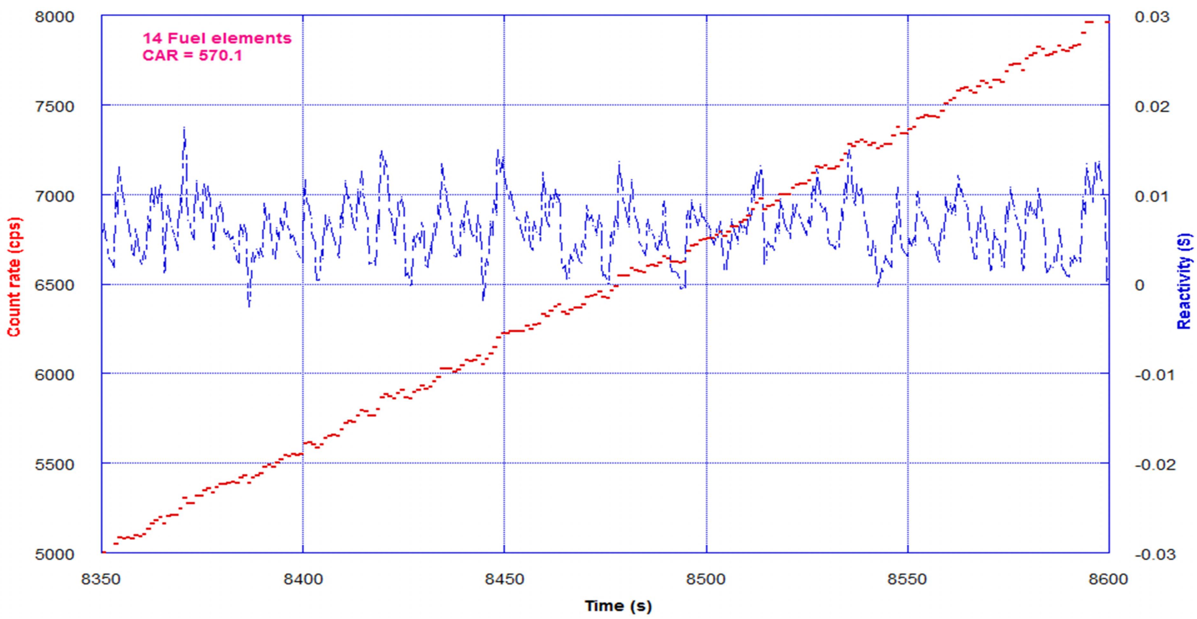

FIG. 3. Count rate of the BF3 detector (cps) as a function of time (s) of the initial core.
NOTE: The initial core is composed of 14 fuel assemblies with a CAR position of 570.1 mm. The reactivity ($) is also shown in the figure.

Measurement of Excess Reactivity

The main objective of this test is to measure the inserted reactivity to the first initial operational core by loading additional fuel assemblies to the minimum critical core [10]. In addition, this test confirms that the shutdown margin for the first cycle core satisfies the design specifications. It is worth mentioning that the reactivity (ρ) is defined in connection with the effective neutron multiplication factor (k_{eff}) as follows:

$$\rho = \frac{(k_{eff} - 1)}{k_{eff}}$$

The fuel assemblies are added to the minimum critical core one by one according to the predetermined fuel loading sequence until the core is fully loaded. Whenever a fuel assembly is added into the core, CARs are withdrawn step by step to approach criticality and 1/M is measured when all CARs are at the same height. The CAR worth, which is a reactivity change caused by a perturbation in a core, is measured from the critical CAR position of the current core to the previous one and hence, the excess reactivity of the new core is determined. The results of this test are presented in Table 3.

TABLE 3. Measured CAR critical position and total worth for each additional FA after reaching the initial criticality.

NOTE: The last column presents the percentage difference between the measured and the simulated CAR worth.

Additional FA, sequence	Measured CAR position (mm)	Total CAR worth ($)	% Diff. from the calculated
Critical core, 14	566.6	0.8958	16.09
FA15,1	454.8	2.4866	14.62
FA16,2	399.4	2.150	13.40
FA17,3	346.1	2.8473	13.09
FA18,4	311.5	2.167	11.85

Measurement of Power Reactivity Coefficient

The objective of this test is to evaluate power coefficient of reactivity by measuring the reactivity variation in response to the reactor power change from zero to full power, as well as during the inverse case [11]. When the reactor power is varied, the reactivity change in response is compensated by the change of critical CAR position. Therefore, the power defect can be determined by the reactivity change, which is measured from the change of critical CAR position. Among other factors, if the power is rapidly raised and then descended after a short time of operation at full power, the change of core temperature with fixed core inlet temperature is the major factor determining the power defect. The core temperature is directly affected by the change of the inlet temperature. To minimize the effect of other factors, the reactor power is raised from zero to full power, as well as during the reverse case as fast as possible. The reactivity change in response to the reactor power variation can be measured by adjusting the inlet temperature [11]. The power reactivity coefficient is defined as the reactivity variation per unit power. For the JRTR case, it can be found from:

$$\frac{\partial\rho}{\partial P}\Delta P = (\rho - \rho_0) - \frac{\partial\rho}{\partial C}\Delta C - \frac{\partial\rho}{\partial X}\Delta X - \frac{\partial\rho}{\partial T}\Delta T$$

where: ρ, ρ_0, P, C, X and T are reactivity, initial reactivity, power, CAR position, Xenon concentration and inlet coolant temperature, respectively.

Fig. 4 qualitatively shows the measured power defect at the indicated power values during the ascending and descending of power. 10 kW is the reference power defect at zero. These plots have been qualitatively constructed based on the experimental plots in graphs 6 and 7 in ref. [12]. As the figures indicate, power defects measured during power descension are larger than those of power ascension; this behavior can be corroborated to more than one reason, where the relatively rapid rise of core inlet temperature during the 5 MW operation can be one of the reasons [12]. However, during this experiment, the measured power coefficients are confirmed negative for the entire power range. However, the uncertainty in the presented data was not discussed. In the present work, the uncertainty is presented in the last column in Table 4, which gives confidence to the measured parameters.

TABLE 4. Measured power defect during power descending and ascending.

NOTE: The data has been reproduced from the plots in Figures 6 and 7 in Ref. 12. The last column presents the uncertainty as derived in this work.

Descending Power		
Power (MW)	Weighted Average of Reactivity Effect ($)	Uncertainty
0.1	-0.00037	±0.00008
1	-0.01001	±0.00076
5	-0.08453	±0.0021
Ascending Power		
3	-0.0425	±0.00095
5	-0.05713	±0.00125

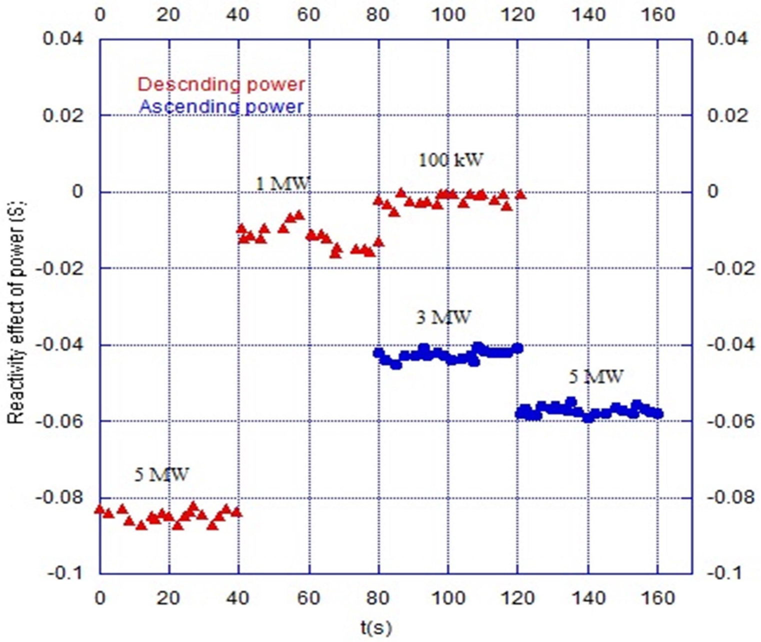

FIG. 4. Qualitative description of the measured power defect as a function of time.

NOTE: The plots represent the measurements during the indicated power values for the ascending and descending of power. The figure has been reproduced from the data presented in Figs. 6 and 7 in Ref. 12.

Thermal Neutron Flux Measurement at IR0

This test is to measure the peak thermal neutron flux at the central irradiation location (IR0) of the JRTR core in order to verify the design criteria. The thermal neutron flux is measured through neutron activation of a cobalt wire contained in a capsule [13]. To perform the irradiation, the capsule is inserted into the expected highest thermal flux position in the IR0 irradiation location. The wires are irradiated for around half an hour when the reactor is operated at the highest nominal power of 5 MW [13]. After irradiation is completed, the reactor is shut down by cutting the electric power for the “loss of normal electric power test”. The irradiated rig is moved to the hot cell to cool off for around one day. The cobalt wires are taken out of the capsules to measure the absolute induced gamma-ray radioactivity. The wires have been cut to smaller pieces in order to measure the activity of each piece separately.

The number of activated ^{60}Co nuclei $N(t_i)$ is calculated from:

$$N(t_i) = e^{-\lambda t_i} \int_0^{t_i} R(t) e^{\lambda t} dt$$

where R(t) is the measured reaction rate, which is proportional to the reactor power.

For the determination of ^{60}Co activity, the 1332.501 keV peak areas have been used. Fig. 5 and Table 5 present the measured thermal neutron flux as a function of distance from the center of the fuel element and the deduced activities from the least-squares fit of the data points. As it is evident, the measured flux at the center of the radioisotope production rig is ~$1.72x10^{14}$ $n/cm^2.s$, which is better than the designed flux of $1.45x10^{14}$ $n/cm^2.s$ [14]. Additionally, the linear fit of the data points gives the value of $1.832x10^{14}$ $n/cm^2.s$ at the center of the core, which is even better than the predicted one. The last column in Table 5 presents the new thermal activities at the points of interest. However, it is obvious that the thermal flux is decreasing as a function of distance from the center of the fuel element.

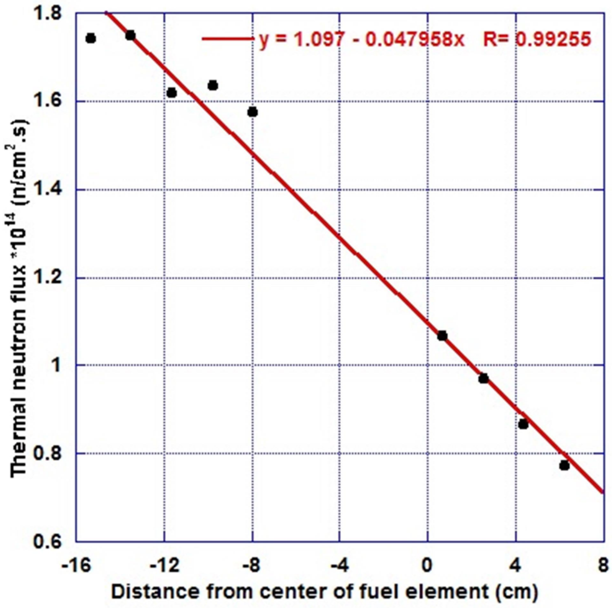

FIG. 5. Deduced thermal neutron flux as a function of distance from the center of the fuel element.
NOTE: The figure represents the measured values as reported in Ref. [14]. The linear fit, the straight line equation and the residue as deduced in the present analysis are also shown.

TABLE 5. Deduced thermal neutron flux as a function of distance from the center of fuel element as reported in Ref. [14].
NOTE: The last column presents the thermal neutron flux as deduced in the present work from the straight line fit.

Distance from vertical center of fuel (cm)	Deduced thermal neutron flux $x10^{14}(n/cm^2.s)$	Thermal flux from linear fit $x10^{14}(n/cm^2.s)$
-15.33	1.743	1.832
-13.49	1.751	1.744
-11.65	1.620	1.656
-9.81	1.636	1.586
-7.97	1.576	1.479
0.67	1.068	1.065
2.51	0.9707	0.977
4.35	0.8674	0.888
6.19	0.7736	0.800
8.03	0.7148	0.712

Neutron Activation Analysis Facility (NAAF) Performance Test

The purpose of this test is to check the performance of the NAAF when the reactor is operating at full power. In particular, the test is designed to verify that the performance of the Pneumatic Transfer Systems (PTSs) and the γ spectrometer meets the design requirements. In addition, the test will enable to generate key data for the operation of the NAAF and demonstrate that actual NAA can be carried out [15]. The test comprises of transferring and retrieving the tested samples using the three PTSs and measuring the time of each process.

An appropriate weight of Standard Reference Material (SRM) samples has been irradiated for sufficient times in the NAA1, NAA2 and NAA3 locations. The analysis was carried out using Ge-based spectroscopy system looking for short, medium and long-lived radioisotopes. The measurement results have been compared with certified/reference values.

The conclusion of this test can be summarized as follows: all three PTS lines function as designed. The gamma spectrometry system also functions well. NAA1 provides a high neutron flux with relatively hard spectrum. NAA2 and NAA3 locations provide well-thermalized neutron spectrum with reasonable flux level for the NAA. The JRTR facility can be used for NAA in JRTR with acceptable accuracy [15].

Radioisotope Production Test

The purpose of this test is to check the performance for the production of ^{192}Ir, ^{99}Mo and ^{131}I isotopes at full power operation. This test verifies the maximum radioactivities of a target capsule for ^{192}Ir, ^{131}I and ^{99}Mo that can be produced at JRTR as proposed [16].

The radioisotope production facility of the JRTR has been designed to be capable of producing more than 2000 Ci of ^{192}Ir every two weeks, 10 Ci of ^{131}I a week and 5 Ci of ^{99}Mo a week when the reactor is operating at full power. The neutron activation was carried on ^{192}Ir discs of 3 mm in diameter and 0.25 mm in thickness. For the production of ^{131}I isotope, TeO_2 target with purity higher than 99.9% was irradiated and for the production of ^{99}Mo isotope, MoO_3 targets were used. The details of material, preparation and irradiation procedures are presented in ref. [16].

During the test, it was possible to produce more than 2716 Ci of ^{192}Ir, 14.54 Ci of ^{131}I and more than 8 Ci for ^{99}Mo. The results of these tests demonstrate that the RI facility works as designed.

Conclusions of Commissioning

The JRTR commissioning plan included three main stages. The tests prior to fuel loading, fuel loading tests and initial criticality tests which include low-power tests. The last stage constitutes power ascension tests and power tests up to the rated full power. All planned experiments have been conducted successfully. These experiments verified the design parameters of the reactor. Particularly, the nominal power, the reactivity feedback, the thermal neutron flux, the radioisotope production facility capability and the performance of the neutron activation facility have been verified to function as designed. Moreover, in some cases, like the thermal neutron flux peak, the radioisotope production capability has exceeded the design prediction. Therefore, the JRTR has been successfully commissioned and is ready to be utilized.

Safety Enhancement of the JRTR in Light of Fukushima-Daiichi Accident

Not like power reactors, the JRTR works under normal temperature and pressure conditions. Therefore, according to the ASMI/ANSI code 51.1 and addenda [17], the Structures, Systems and Components (SSCs) of Safety Class-1 SC-1 and SC-2 are not applicable to the JRTR. Hence, only SC-3 is applicable to the JRTR. For the JRTR, the main nuclear safety functions of the SSC are to [17]:

- Provide secondary containment for the radioactive material holdup, isolation or heat removal with high reliability;
- Remove radioactive material from the atmosphere of confined spaces outside primary containment containing SC-3 equipment;
- Provide or maintain sufficient reactor coolant inventory for core cooling;
- Maintain geometry within the reactor to ensure core reactivity control or core cooling capability;
- Structurally load-bear or protect SC-3 equipment;
- Provide radiation shielding for the control room or offsite personnel;
- Ensure nuclear safety functions provided by SC-3 equipment (e.g., heat removal or provide lubricant for pumps and heat exchangers);
- Provide actuation or motive power for SC-3 equipment;
- Provide information or control to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-3;
- Provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety functions required of SC-3 equipment;

- Provide an acceptable environment for SC-3 equipment and operating personnel.

Safety Classification System of the JRTR

The safety classification of the JRTR is based on the classification system presented in Ref. [17]. The SC-3 or NNS classes have been implemented in the JRTR design and relied upon to accomplish nuclear safety functions. The Non-Nuclear Safety (NNS) is defined for equipment not included in any of the SC-3 equipment and is not relied upon to perform a nuclear safety function.

The quality class is designated to design, fabricate, install and test the safety related to the SSCs in accordance with the standards that are appropriate for their intended safety function. Quality classification is generally consistent with safety classification. The quality classification of the SSCs depends on the ASME NQA-1 that classifies the quality to Q, T or S classes [18]. It is worth mentioning that all SC-3 components have been classified as Q class, Seismic Category-I and Class-1E as an applicable electric class.

For SC-3 components, quality assurance program requirements in ANSI NQA-1 [18], or another equivalent program, are applied. Quality class T is applied to SSCs whose functioning is essential for the normal operation of the reactor, or the failure of which could affect the reliability of the safety class equipment. For this quality class T, selected QA program requirements of quality class Q or QA program requirements of applicable codes and standards are implemented. Quality class S is applied to all SSCs that are not classified as quality class Q or T.

Three seismic categories have been adopted for the SSCs that are essential for the safety of the reactor. SSCs that are required to maintain their integrity and function during and after Safe Shutdown Earthquake (SSE) are categorized as Seismic Category I. SSC components that are required to maintain their structural integrity under load induced by SSE are categorized as Seismic II. Seismic Category III includes SSCs that are not included in either category I or II.

The SSCs that are related to safety functions are classified as electrical class (1E) or electrical non-class 1E (Non-1E). These terms are defined in IEEE Std. 100 [19]. Table 6 presents a partial list of the safety classification system of the JRTR. The table introduces the safety, seismic, quality and electrical classes of the main SSCs.

TABLE 6: Partial list of JRTR SSCs according to safety class, seismic category, quality class and electric class.

System	Safety Class	Seismic Category	Quality Class	Electric Class
Reactor Building Structure	3	I	Q	NA
Reactor Concrete Island	3	I	Q	NA
Reactor Pool Liner	3	I	Q	NA
Fuel Assembly	NA	I	Q	NA
Reactor Structure Assembly	3	I	Q	NA
CRDM	3	I	Q	NA
Spent Fuel Storage Rack	NNS	I	Q	NA
Primary Cooling System	3	I	Q	NA
Secondary Cooling System	NNS	Non	S	NA
Alternative Protection System	NNS	Non	T	Non-1E

Enhancement of the SSCs of the JRTR

After Fukushima Daiichi accident, which was classified by the Japanese Nuclear and Industrial Agency to level 7 at the International Nuclear Event Scale [20], the IAEA revised the safety standards to enhance the safety of nuclear installations. The revised safety standards can be found in the publications of the IAEA. These revised standards can be featured by the following:

A. Preventing unacceptable radiological consequences to the general public and environment (Criteria for Beyond Design Basis Accident);
B. Preventing long-term off-site contamination (alleviate severe accident);
C. Preventing severe accidents and reinforcing design bases.

The JRTR, which was under construction during the Fukushima-Daiichi accident, has been affected by the new international safety standards and norms. However, it is worth to note that the JRTR site is far from the sea shore and cannot be affected by a tsunami. In addition, the core of the reactor is always under a sufficiently large pool of demineralized water compared to the generated heat and always safely cooled naturally and therefore, a similar accident to the Fukushima-Daiichi accident cannot occur. Nevertheless, all recommendations and lessons learned from the Fukushima-Daiichi accident have been adopted and implemented to improve the safety of the JRTR. In the following section, there are examples on the improvements that have been implemented to the JRTR components.

Alternative Protection System (APS)

In the Preliminary Safety Analysis Report (PSAR) [21], the APS was classified as a Non-Nuclear Safety system. However, it is described in the PSAR to act as a diverse protection system to perform prevention and mitigation of anticipated transient without scram. The APS also mitigates the effect of the failure of the Reactor Protection System (RPS). Therefore, the APS has been upgraded to an item important to safety, unlike the original design.

Automatic Seismic Trip System (ASTS)

In the JRTR, the ASTS is the only system that is responsible of safely tripping the reactor, which is in accordance with the NS-R-4 safety standards on the Postulated Initiating Events (PIEs). For this purpose, the system is equipped with 4 seismic sensors and the trip logic is 2 out of 4. Therefore, the seismic monitoring system has been upgraded in terms of the quality class of hardware and software. In addition, a function generating the automatic seismic trip signal when a seismic motion exceeds OBE (Operation Bases Earthquake) has been added. Moreover, the system has been classified as T-class, seismic category I and Non-1E. Additional UPS also has been built in the cabinet so as to store the earthquake-related data.

Emergency Water Supply System (EWSS)

The EWSS is designed to cover the reactor core with water when multiple ruptures of a beam tube occur in order to cool the core for a sufficient period of time. The EWSS injects, by gravity, the demineralized water from the Demineralized Water Supply Tank (DWST), into the reactor outlet Primary Cooling System (PCS) pipe by opening two parallel Motor Operated Valves (MOVs). Since the multiple ruptures of a beam tube were classified as BDBA, the EWSS was originally classified as a non-nuclear safety system. However, the injection line of the EWSS penetrates the reactor pool liner and is connected to the reactor outlet PCS pipe, which is classified as safety class 3. Since the structural integrity of the injection line part that penetrates the pool liner must be maintained during the motion of the MOVs, the portion of the system between the flow orifice outside the reactor pool and the injection nozzle inside the reactor pool has been classified as a safety class 3 in accordance with case 6(c) of Sec. 3.3.2 Safety Class Interfaces of ANSI 51.1 and has been upgraded to seismic category I, including the MOVs.

Radiation Monitoring System (RMS)

In open-pool type research reactors, most of the critical radiation accidents are closely related to the pool and connected to the primary cooling system. Therefore, the JRTR has been equipped with RMS.The RMS components that are classified to SC-3 are the Reactor Gamma Monitoring System (RGMS), the PCS neutron monitoring system, the PCS gamma monitoring system and the pool radiation monitoring system. The RMS also has been enhanced by adding general-purpose RMS channels, for local radiation dose rate locations that are routinely occupied by operating personnel and other places where changes in radiation levels may occur. These have been classified as NNS and quality class T.

Pool Liner Integrity Enhancement

The stainless steel liner plate of the reactor pool covers the entire internal surface of the pool. The main function of this liner plate is to provide a leak-tight barrier against any possible leakage of pool water. It is worth mentioning that the non-destructive test during the fabrication and installation of pool liner was not possible due to technical reasons. Therefore, the internal integrity of the welded joints has been confirmed by the following additional tests:

- In addition to their qualification tests in accordance with the fabrication procedure, the welders were subjected to practical tests and accordingly certified. Also, weld verification testing was doubly performed.
- For each welding posture of welding angle, a test sample was attached to the actual welding part and welded together with the original part in sequence. Then, the sample was tested to verify integrity.

Air Discharge System (ADS)

In all conditions, the ADS keeps the pressure inside the reactor building negative compared to the outside, so that the air leakage from the reactor building is prevented. In case of emergency, the purpose of the system is to reduce the public dose and to give the management the proper tools to control the path of the released gases [22]. The main components of the system are the building, the filter train, the exhaust fan, the isolation valve, RMS for noble gas, … etc. Since the ADS is installed in a separate building outside the reactor building and the system should not affect the safety features of the reactor building, safety class (SC-3) valves have been installed at the reactor building penetrations for air exhaust and these are kept closed during normal conditions. Thereby radioactive material leakage to the environment is reduced and monitored. The exhaust duct is also connected to the reactor stack. Additionally, the ADS building and piping have been upgraded from Seismic class II to I. All other components of the system are classified as NNS, T and II for safety class, quality class and seismic class, respectively. HEPA filter and activated carbon filter are included in the system and hence, the majority of particles and halogens can be filtered out. Because most of the released radioactive material is noble gas, a noble gas radiation monitor has been installed.

Two Mobile Diesel Generators

In addition to the main, 1000 kVA diesel generator, two mobile diesel generators for supplying the necessary power to the load under the emergency situation have been added to the plant. These are with an output of 300 kVA and are accommodated in the ADS building, which is independent of the reactor and is designed to a seismic category I, having sufficient space and an independent entrance for maneuvering the generators.

Additional Upgrades

Other components and equipment have been added to the facility. To accommodate these additions, the total area of the facility has been increased. Summary of the major additions to the building and systems can be found in Table 7.

TABLE 7. Summary of major additions to the JRTR buildings and systems.

Description		Original Proposal	Additional Upgrades
Buildings	Reactor Building	6-story building for the reactor confinement	- Reactor confinement was enlarged from 2,100 m^2 to 2,260 m^2. - Additional air lock door.
	Service Building	5-story building consisting of auxiliary areas for operation and RI production	Auxiliary areas were enlarged from 4,700 m^2 to 5,600 m^2.
	ADS Building		2-story building containing air discharge system (ADS) and two mobile diesel generators.
	Cooling Tower and Others	- 1 cooling tower with 3 fans - 1 pump house - 1 diesel generator building - 1 fire water tank - 1 stack structure	- 5 closed-type mechanical cooling towers. - 2 Fire water tanks. - 4,700 ton water storage tank.

Description		Original Proposal	Additional Upgrades
Systems	Reactor System	- Reactor core and core containing structures - Reactor Protection System - Seismic Monitoring System - Reactor Control and Monitoring System - Radiation monitoring system, … etc.	- Automatic Seismic Trip System (ASTS). - Quality class upgrading of I&C.
	Primary Cooling System and Connected Systems		- Increase of Decay Tank capacity. - Position switches on flap valves. - Siphon break valves. - Reactor pool platform with guide tube. - Pool liner integrity enhancement.
	Supporting Systems	- Electrical System Including Emergency Power Supply System - Fire Protection System - Communication Systems - Lighting Systems - Compressed air system	- 2 mobile diesel generators. - Additional field instruments. - Enhanced air compressor.
	HVAC	- Reactor building HVAC - RI building HVAC	Enlarged HVAC and firefighting system. - Air discharge system.
	Others	Fuel and reactor component handling and storage system - Radiation shielding - Radwaste management systems	Additional seismic support for spent fuel storage rack. - Enhanced lifting utilities. - Additional elevator. - Enhanced spent resin handling. - Enhanced fresh resin handling. - Additional engineering.

Conclusion

Following the Fukushima-Daiichi accident, extensive studies and discussions between JAEC, KDC, Regulator, the IAEA and other consultants have been carried out reassessing the safety features of the JRTR and making sure that the facility implements the lessons learned from the accident. As a result, several SSCs have been reclassified and upgraded in terms of either safety class, quality class or seismic class. Therefore, the JRTR and the associated facilities are safe during all anticipated operational conditions. In light of these upgrades, the facility has been successfully commissioned.

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