Study of structure systems and components classification of Reaktor Daya Eksperimental - RDE based on life cycle management

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Abstract. Reaktor Daya Eksperimental (RDE) is a pebble-fueled, gas-cooled non-commercial power reactor having thermal power of 10 MW that will be constructed in PUSPIPTEK Serpong. The objective of the small-scale power reactor is to demonstrate to general public the utilization and safety features of nuclear power reactor. Aging management of a nuclear installation should have been considered since the design, construction, commissioning phase and operation phase as well. It is expected that, in designing structure, system, and components (SSC) of a nuclear installation, material quality selection, component design life cycle, surveillance plan, and maintenance schedule in accordance with its purpose should be carried out carefully. As a consequence, if this SSC fails, it can give significant impact to environment. Classification of component of RDE planned by Indonesia is not available yet. Therefore, this paper describes RDE component classification with regard to life cycle management. SSC classification is carried out based on IAEA’s Safety Guide, and regulations in effect in Indonesia as well as life cycle management. The screening of SSC candidates that have been identified based on aging object easiness of refurbishment, it's found that there are eight critical SSCs of RDE. The reactor core and internals are the critical SSC that has the most critical components of aging mechanism assessment.

Keywords: RDE, SSC, classification, life cycle management, ageing management

1. Introduction
Indonesia has planned to develop a high-temperature gas-cooled reactor (HTGR), which is called Reaktor Daya Eksperimental (RDE). This reactor is a pebble-fueled, gas-cooled non-commercial power reactor having thermal power of 10 MW that will be constructed in PUSPIPTEK Serpong. This reactor is managed and will be operated by National Nuclear Energy Agency of Indonesia (BATAN). The objective of the small-modular power reactor is to demonstrate to general public the utilization of the reactor producing energy [1]. The selection of this high-temperature gas-cooled reactor type is based on several aspects, one of which is the failure of the water-cooled reactor in Fukushima Daiichi in Japan in 2011 [2–4]. The HTGR has inherent safety features, such as negative reactivity feedback coefficient, high heat capacity of the core, and inert helium gas as coolant. These will ensure sufficient safety margin of the reactor, without much dependence on active safety system [5–6].
Therefore, the HTGR designed has remarkable economic advantages, because the same safety level required for the existing reactors can be achieved by using a simple passive safety system. Passive safety systems fulfill the basic safety functions, and have design characteristics such as large temperature reactivity feedback coefficients, to accommodate reactivity insertion without causing fuel damage. The lower reactivity design and lower power density, as well as the high thermal capacitance of the reactor core structure, causing the maximum temperature of the fuel to remain under the design limit of the structure, thus the fuel temperature is below the limit of particle degradation. It is also supported by a robust fuel design that prevents the release of fission products. In addition to fulfilling the safety design philosophy of the established gas-cooled reactor, the design features should be considered appropriately to make such reactors interesting in the future. Moreover, aging management of a nuclear installation should have been considered since the design, construction, commissioning phase and operation phase. It is expected that, in the designing structure, system, and components of a nuclear installation, material quality selection, component design life cycle, surveillance plan, and maintenance schedule in accordance with its purpose should be carried out carefully.

The most crucial safety principle in a nuclear reactor facility is capability for the reactor to be shut down and cooled safely following the postulated accidents. To meet this principle, the structure, system, and components (SSC) important to safety should be designed, developed, constructed, and tested based on the quality standards related to the importance of their safety functions. SSC should be classified in accordance with their safety importance and/or significance using proper classification guidance.

Japanese experience in designing High Temperature Reactor (HTR) safety, adopted by China for HTR-PM and Tsinghua University, and IAEA’s Safety Guide that describing components classification have become important lesson learned for the preparation of RDE SSC classification [7-8]. Tae Ryong-Kim (2016) conducted component classification for research reactor based on IAEA’s regulation and South Korea’s regulatory body [9]. Prabir C. Basu (2017) reviewed the available references and summarized the development of PSA method to derive nuclear power plant component fragility towards external hazards [10]. Meanwhile, Michael Türschmann (2015) performed a research and development on internal hazards. His study carried out considers all potential dependencies (impact dependency on different hazards, dependency on safety function required for controlling hazard consequences that initiate the event and dependency on the risk caused by structure failure, system, and components), in the plant quantification model. This research concludes that compartment failure probability due to fire for each compartment should be mapped as a requirement that component and cable located in the compartment cannot be operated because of fire [11].

SSC classification is closely related to aging management planning. The basic concept of aging management is how to overcome safety-related degradation and to take corrective measure before the loss of SSC integrity and functional capability [12]. Effective aging management is in practice carried out under the coordination with the existing program, including maintenance, in-service inspection and surveillance, operation, technical support program, and external program, such as research and development. Therefore, an effective aging management is required at all SSC phase using a systematic approach to manage aging. Aging management should provide a framework to coordinate all programs and activities related to understanding, controlling, monitoring, and mitigating aging effect of component and structure installation [13]. Classification of SSC should fulfill reactor safety philosophy, where the RDE design should be developed such that it meets the defense in depth and multi-barrier principle. Therefore, all safety measures should be provided more than one type of defense, so that individual component failure does not give consequences to reactor safety [14]. This design should also meet safety function, which allows the reactor to be shut down either under operational condition or accidental event. The residual heat after reactor shutdown under the operational or accidental condition should be able to be removed [15].

Based on those various reviews, classification of component of RDE planned by Indonesia is not available yet. In order to obtain a classified SSC RDE classification in asset management, this paper
describes RDE component classification with regard to the IAEA safety guide and life cycle management, regulations in effect in Indonesia, and reactor safety design consideration.

2. Methodology

2.1. SSC’s classification based on IAEA safety guide

Aging is a natural process as indicated by the degradation of SSC capability to perform its function following age or time function. This process eventually causes material degradation and then lower or even eliminate SSC essential capabilities for safety and function in accordance with the required criteria. The critical SSC is component that becomes part of safety system and/or if it fails or experiences malfunction, radiation exposure to workers or people may occur. The critical SSC should fulfill the following criteria: not redundant, not easily repaired, and not easily replaced. Facility safety and utilization will be interrupted unless preventive and/or corrective measures are taken. Safety Class of SSC is categorized into sub-classes as shown in Table 1.

| Category | No. of Sub-Class | Sub-Class |
|----------|------------------|-----------|
| Class A: | 1) Equipment/component if used/operated will relate/affect directly to safety; |
| Class B: | 2) Equipment/component if used/operated will relate/affect indirectly to safety; |
| Class C: | 3) Equipment/component if used/operated will not relate/affect directly to safety; |

Classification method of component important to safety is given in IAEA SSG No. 30 on SSC classification for NPP. In safety standard, the classification methods of NPP component—starting from preparation of list of all components, evaluation of SSC critical to safety, evaluation of the selected SSC and preparation of list of the selected SSC into a generic group, equipment/component types, material, service condition and component degradation—are described, as shown in Figure 1.

![Figure 1. SSC's classification methodology [17]](image-url)
Component classification for NPP in Indonesia is regulated by the regulatory body, BAPETEN. Its regulation is pursuant to the IAEA Safety Guide. In conjunction with the RDE development plan, BAPETEN has issued regulations on the list of design information [18] and operating limiting conditions of the power plant [19]. There is no BAPETEN Chairman Regulation that specifically regulates SSC classification for HTGR. The available regulation is for light water reactors, such as BWR. Similar condition of component classification occurs to the IAEA Safety Guide. While performing identification, licensee should take into account the following consideration:

- Parameter of plant operation;
- The results of research and development on aging;
- Critical SSC;
- Material, component, and structure;
- Experience on the aging of critical SSC relevant to nuclear and non-nuclear industries;
- Testing requirements;
- Maintenance requirements; and
- Operation period that has been estimated during pre-operation.

Identification is performed from construction to operation phase.

### 2.2. Asset Life Cycle and Asset Management

Two important aspects that should be considered in aging management are asset life cycle and asset management. Asset life cycle includes planning, operation, maintenance, and decommissioning phase. Asset management is needed to ensure that a plant can economically operate as its expected life cycle [20]. RDE aging management is carried out based on graded approach according to its system complexity, function, and strategy. The strategy of aging management is aimed to ensure the SSC integrity and performance required by implementing a systemic aging management process, including detection and prevention of SSC degradation process. Identification of potential weakness due to unavailability of information is carried out to anticipate inability to understanding and estimating aging at design and construction stage of RDE; and early aging. Therefore, the RDE life cycle management should be considered as a single period that covers all stages, starting from design to decommissioning, as shown in Figure 2.

RDE is designed and constructed using appropriate code and standard, procedure, and high quality material, and is inspected/tested comprehensively. RDE operation is defined at design stage and can be extended in operation stage if supported and justified for the extension of its operational life cycle. This operational life cycle will determine the total amount of the generated electricity by RDE. The life cycle management of RDE is an integration of safety management, aging management, and business management, as well as economic considerations during its life cycle. This management is aimed to:

- Maintain RDE level of performance, including its safety.
- Optimize its operation, maintenance, and service life cycle of critical SSC.
- Maximize the return rate of investment during RDE operational life cycle.
- Take into account national strategy to finance the RDE life cycle management including decommissioning, fuel management, and waste management.

The basic process developed for the RDE project is meant for technical and economic evaluation required at planning stage of life cycle management. Figure 4 shows a description on the process stage of the life cycle management planning. The simplified flow chart indicates the main steps needed in the development of life cycle management planning for identification of the candidate of critical SSC. These steps are performed sequentially in accordance with the criteria in the diagram on the candidate of critical SSC.
2.3. RDE Design
RDE is planned to generate electricity and cogeneration, several studies on fuel loading schemes with the OTTO fuelling cycle on HTR-PM [22], thermal fuel characteristics of pebble bed on HTGR [23,24], turbine power conversion system using helium Brayton cycle design on HTR [25] and helium purification systems on RDE have been analyzed [26]. It gives an overview of the design flow diagram RDE. This 10 MW RDE has pebble fuel and high-temperature gas coolant as well as cogeneration with operational data as shown in Table 2 and Figure 3. Helium gas from a compressor cools the reactor core with temperature of 250°C and pressure of 3 MPa. Helium gas removes heat of 10 MWth and flows out at temperature of 700°C and pressure of 3 MPa and flow rate of 4.3 kg/s.

Figure 2. *Life Cycle* management steps of critical SSC [21].

Figure 3. Flow diagram of RDE
Heat is transferred to cooling water in the steam generator whose water is obtained from the feed water pump at 160°C and flow rate of 4 kg/s. The steam temperature at the outlet is 530°C and the pressure is 6 MPa. Heat is utilized to drive the turbine generator of 2 stage to be used for cogeneration. Residual heat from the turbine having high temperature is condensed for feed water of steam generator. Meanwhile, water that removes the condenser heat discharges its heat to a cooling tower.

**Table 2. Main data of RDE Design [27]**

| Parameter                              | RDE     |
|----------------------------------------|---------|
| Inlet temperature He gas, °C           | 700     |
| Outlet temperature He gas, °C          | 245     |
| Pressure He gas, MPa                   | 3.0     |
| Steam temperature, °C                  | 530     |
| Water inlet temperature, °C            | 160     |
| Steam outlet pressure, MPa             | 6.0     |
| Feed water pressure, MPa               | -       |
| Thermal power, MWt                     | 10.0    |

**Steam Generator**

| Parameter                              | RDE     |
|----------------------------------------|---------|
| Primary coolant mass flow of He, kg/s  | 4.4     |
| Mass flow rate of steam, kg/s          | 4.0     |
| Number of tubes                        | 93      |
| Tube outside diameter (OD), mm         | 23      |
| Heat transfer area, m²                 | 70      |
| Vessel diameter, m                     | 1.5     |
| Vessel height, m                       | 12.35   |

3. Results and Discussion

Screening is performed at construction stage based on the predefined method. In accordance with the BAPETEN Chairman Regulation No. 7 Year of 2012, as an alternative to PSA, the method used should be justified and documented. The selected screening method refers to the IAEA’s recommendation and BAPETEN Chairman Regulation, based on the level of effect of SSC to safety; and the level of easiness of SSC reparation or refurbishment.

SSC screening method consists of 3 steps. The first step is the preparation of a list for critical safety-related SSC. The second step is further evaluation of SSC resulted from the first step in order to determine elements that, in case of failure, can deteriorate or diminish (directly or indirectly) their safety function. In the third step, the SSC resulted from the second step is evaluated further to identify elements that might cause SSC failure and to justify components not included in the SSC.

3.1. Classification of RDE-SSC

In the classification of structures, systems and components of experimental power reactors, listings and grouping of components have been performed based on data obtained from potential vendors. There are 10 SSC's groupings that have been compiled based on the importance of safety and easiness of refurbishment as shown in Table 3. From those aspect there are four category of a component stated, i.e.: category A when it is very difficult, B is technically difficult and costly, C normal and D easily refurbishment. A component in the category very difficult - A, due to the location of these components are in the internals of the reactor or directly affected by exposure to radiation. Components in the category of very difficult have high qualifications, and is designed to be operated during, or even exceed the life of the reactor. Therefore, it is not expected to replace the component during the life of the reactor.
The number of components in the RDE are huge, therefore not all of components are shown in Table 3, only the main components are listed. As a consequence in one group there are different categories of easiness of refurbishment components. Beside that in the first step of screening, categorization is done whether the SSC important to safety or not.

Table 3. SSC screening based on safety and easiness of refurbishment

| Group No. | SSC                      | Components                           | Number of components | Important for safety | Easiness of refurbishment |
|-----------|--------------------------|--------------------------------------|----------------------|----------------------|---------------------------|
|           | Reactor System           | Reactor vessel assembly              | 22                   | Yes                  | A / B                     |
|           |                          | Hot gas duct                         | 8                    | Yes                  | A                         |
|           |                          | Steam Generator                      | 20                   | Yes                  | A/B/C                     |
|           | Group 2. Confinement    | Reactor Building and systems         | 6                    | Yes                  | B                         |
|           | Group 3. Helium         | Sistem Pemurnian Helium Train 2, (Post Accident) | 19                   | No                   | C / D                     |
|           |                          | Sistem Pemurnian Helium Train 1, (Normal Operation) | 16                   | No                   | C / D                     |
|           |                          | Helium Supply System,                | 7                    | No                   | C / D                     |
|           |                          | Dump System Helium,                  | 6                    | No                   | C / D                     |
|           |                          | Gas Evacuation System,               | 10                   | No                   | C / D                     |
|           | Fuel handling & storage | Fuel Feed equipment                  | 9                    | No / Yes             | B / C / D                 |
|           |                          | Fuel discharge equipment             | 15                   | Yes                  | B / C                     |
|           |                          | Fuel storage                         | 3                    | Yes                  | C                         |
|           | Group 5. Water/steam     | All.                                 | 37                   | No                   | A / B / C                 |
|           | Group 6. Cooling water   | All.                                 | 30                   | No                   | B / C                     |
|           | Group 7. Radioactive     | Liquid Waste System                  | 13                   | Yes                  | C                         |
|           |                          | Solids storage                       | 1                    | Yes                  | C                         |
|           |                          | Decontamination equipment            | 21                   | Yes                  | C                         |
|           |                          | Water Extraction System for Helium Supporting Systems | 7                    | Yes                  | C                         |
|           | I & C                    | Supervisory komputer                 | 36                   | Yes                  | B                         |
|           |                          | RPS Panel                            | Yes                  | B                     |
|           |                          | Neutron flux instrumentation:        |                      |                      |                           |
|           |                          | flux sensor                          |                      |                      |                           |
|           |                          | Instrumentation channel with         |                      |                      |                           |
|           |                          | indicator for remote shutdown        |                      |                      |                           |
|           |                          | Hot gas sensor temperatur            |                      |                      |                           |
|           |                          | Detector, filter and preamp included |                      |                      |                           |
|           |                          | Accident monitoring system          |                      |                      |                           |
|           | Group 9. Electrical      | Main distribution board              | 25                   | No                   | B / C / D                 |
|           |                          | Emergency diesel aggregates         |                      | Yes                  | B                         |
|           |                          | Uninterruptible power supply         |                      | Yes                  | C                         |
|           |                          | For 220 VDC                          |                      |                      |                           |
|           | Group 10. Civil structure| Reaktor Building,                    | 4                    | Yes                  | A                         |
|           |                          | Reactor Auxiliary Building,          | 44                   | Yes                  | B                         |

Aging analysis in RDE is aimed to identify the aging mechanism or degradation mechanism and increase in failure probability due to aging, to ensure structural integrity, and to predict the rest of life cycle of the structural component. Table 3 shows data SSC candidates based on the Work Breakdown Structure (WBS) of RDE. In the second phase of screening of the SSC, where perform the rating major
components that impact on safety. The main data obtained from the screening results of the SSC candidate that have been identified based on the aging objects and refurbishment easiness include eight critical SSCs for RDE, i.e.:

1. Reactor pressure vessel and relative equipment (JAA/1.1.1)
2. Reactor core and its internals (JAA/1.1.2)
3. Control rods and their mechanism (JAA/1.1.3)
4. Small ball shutdown unit (JAA/1.1.4)
5. Hot gas duct (JEC/1.2)
6. Blower (JEA/1.3.1)
7. Steam Generator pressure vessel and relative equipment (JEA/1.3.2)
8. Reactor Building (UJA)

Critical SSC for RDE provides perspective on the reason of making the SSC candidate in the main data. The results of this section include matrix format for each critical SSC to identify aging mechanism, information of potential causes, and the impact ranking of this critical SSC.

The safety-related critical SSCs are structure, system, and component that become parts of a safety system and/or structure, system, and component, if failed or malfunctioned, causing radiation exposure to workers or people. Meanwhile, the critical SSC includes component important to safety and vulnerable to aging. The critical SSC should meet the following criteria: not redundant; not easily repaired; and not easily replaced. If this SSC fails, it can give significant impact to environment.

Since aging affects reactor safety, detection of degradation due to aging and assessment on the SSC aging to safety are needed. RDE has a large number of components and types of SSC. Therefore, SSC screening based on safety should be carried out. In accordance with SSG-10, the definition of the safety-related critical SSC is SSC that has safety functions. The safety functions are to:

a. Shut the reactor down and maintain it in safe state both for operation and design basis accident;

b. Remove heat, especially heat of the core of the design basis accident, after the reactor has been shut down;

c. Contain radioactive material in order to prevent or limit unexpected release to environment

This safety-related critical SSC needs to be assessed in term of environment and service condition, aging mechanism, aging mitigation, and detection of aging degradation. The results of this detection are evaluated to ensure continuous and safe operation and to perform refurbishment of SSC experiencing degradation before failure occurs. Then, SSC screening based on easiness of refurbishment or reparation is carried out since its reparation or refurbishment requires specific effort and time, so that a proper planning should be prepared.

Safety-related critical SSC and its very difficult refurbishment will be the subject of aging management. Data obtained from SSC testing and monitoring of aging management objects are collected and are used to estimate the trend in order to predict the remaining life cycle. The design and specification evaluation results related to SSC aging are shown in the table of RDE System Component Aging Mechanism. The Document of RDE Aging Management is prepared based on the effect of SSC toward safety, easiness of refurbishment or reparation of SSC, material, minimization, service condition, surveillance, and mitigation.

Figure 4 shows a diagram between SSC and assessment index based on easiness of refurbishment to determine the critical SSC ranking. It can be seen in Figure 4 that “reactor core and internal (JAA/1.1.2)” is SSC with screening sequence that is very difficult to be refurbished among eight critical SSC. Meanwhile, “reactor building (UJA)” is critical SSC that has easiness of refurbishment with easy category. Special attention should be given to ensure proper technical margin. This margin consists of design margin, operational margin, and safety margin to prevent component failure. In this document, screening and identification of SSC are only based on design margin (in this case, conceptual design document of WBS RDE). This margin should cover operational margin that considers the impact of wear out and degradation mechanism related to operational life cycle to ensure
safety and function of each component during the RDE operation. Potential aging mechanism that affect safety and SSC function during design life cycle include: brittleness due to heat and radiation, fatigue, corrosion, crack-initiating environment, creep, and wear out.

Figure 4. Critical rank of SSCs after screening based on easiness of refurbishment.

Figure 5. Aging mechanism diagram on critical SSC of RDE [21].

Figure 5 shows the identification results in form of diagrams of aging mechanism that occurs on the critical SSC component for RDE. It can be seen from Figure 4 that “reactor core and internal (JAA/1.1.2)” is critical SSC of RDE that has the most aging mechanism, i.e. changes in properties such as creep, motion, fatigue, wear, and erosion. Figure 4 also indicates the identification results with changes in properties due to radiation and temperature affecting the critical SSC, including “reactor building (UJA)”. In addition, there is also creep due to stress. On the other hand, “reactor pressure vessel and relative equipment (JAA/1.1.1)” and “reactor building (UJA)” does not experience creep, then there is no changes in geometry, such as crack or rupture.

3.2. Classification of RDE-SSC based on life cycle method

The components screening based on life cycle method are needed to assess whether the SSC RDE qualify as a critical component. The screening steps are:

1. Critical SSC have important safety risk
2. SSC is critical for electricity production and can cause scram black out
3. Critical SSC failure is very expensive
4. Critical SSC failure can cause regulatory issues
5. The critical SSC is experiencing significant degradation
6. Critical SSC spare parts are not available
7. SSC is critical with chronic maintenance issues

Table 4. SSC screening based on life cycle method

| Group No. | SSC                  | Screening Steps | Criteria performance |
|-----------|----------------------|----------------|---------------------|
|           |                      | 1   | 2   | 3   | 4   | 5   | 6   | 7   |
| Group 1   | Reactor System       | Yes | Yes | Yes | Yes | Yes | Yes | Yes | 7   |
| Group 2   | Confinement          | Yes | No  | Yes | No  | Yes | Yes | 5   |
| Group 3   | Helium purification  | Yes | No  | No  | Yes | No  | No  | 2   |
| Group 4   | Fuel handling & storage | Yes | No  | No  | Yes | No  | Yes | 4   |
| Group 5   | Water/steam cycle system | Yes | No  | Yes | No  | Yes | Yes | 5   |
| Group 6   | Cooling water system | Yes | No  | No  | No  | Yes | Yes | 3   |
| Group 7   | Radioactive waste processing system | Yes | No  | Yes | No  | Yes | No  | 4   |
| Group 8   | I & C                | Yes | Yes | Yes | Yes | No  | Yes | 6   |
| Group 9   | Electrical           | Yes | Yes | No  | No  | Yes | No  | 3   |
| Group 10  | Civil structure       | Yes | No  | No  | No  | No  | No  | 1   |
The screening results in Table 4 show that SSC of the reactor system is the most critical SSCs, have important safety risk and can cause scram black out, it means can interrupted of electrical power supply. The SSCs meet all screening criteria by life cycle management methods, therefore it can be categorized as SSC candidate critical component of type A. While civil structure is the opposite, is a SSC that does not meet the critical component category, thus it becomes candidate as SSC type B, C, or D.

SSC’s RDE screening has been carried out based on the IAEA safety guide SSG-2.12 concerning Ageing Management for NPP, and the BAPETEN chairman Regulation No. 7/2012 on Aging management of non nuclear reactor installations [28]. Moreover it also conducted screening of components based on life cycle management method, which aims to synchronize between aging management and asset management, system assets, asset performance, risk, and financing SSC in its life cycle. Both of these screening methods should be used, as well as to meet the aging management, is also required in the manufacturing process, implementation and improvement of SSC RDE in a sustainable manner.

4. Conclusion
Method of classification and documentation of SSC data and information carried out has been in line with the RDE management system, by considering safety function, service location and condition as well as life cycle management. Based on the results of SSC screening, there are 8 critical SSC components of RDE. The reactor core and internals are the critical SSC that has the most critical components of aging mechanism assessment.

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