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UPGRADING THE ELETRICAL SYSTEM OF THE IEA-R1 REACTOR TO AVOID TRIGGERING EVENT OF ACCIDENTS

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ABSTRACT

The IEA-R1 research reactor at the Institute of Energy and Nuclear Research (IPEN) is a research reactor open pool type, built and designed by the American firm "Babcox & Wilcox," having as coolant and moderator demineralized light water and Beryllium and graphite, as reflectors. The power supply system is designed to meet the electricity demand required by the loads of the reactor (Security systems and systems not related to security) in different situations the plant can meet, such as during startup, normal operation at power, shutdown, maintenance, exchange of fuel elements and accident situations. Studies have been done on possible accident initiating events and deterministic techniques were applied to assess the consequences of such incidents. Thus, the methods used to identify and select the accident initiating events, the methods of analysis of accidents, including sequence of events, transient analysis and radiological consequences, have been described. Finally, acceptance criteria of radiological doses are described. Only a brief summary of the item concerning loss of electrical power will be presented. The loss of normal electrical power at the IEA-R1 reactor is very common. In the case of Electric External Power Loss, at the IEA-R1 reactor building, there may be different sequences of events, as described below. When the supply of external energy in the IEA-R1 facility fails, the Electrical Distribution Vital System, consisting of 4 (four) generators type "UPS", starts operation, immediately and it will continue supplying power to the reactor control table, core cooling system and other security systems. To contribute to security, in the electric power failure, starts to operate the Emergency Cooling System (SRE). SRE has the function of removing residual heat from the core to prevent the melting of fuel elements in the event of loss of refrigerant to the core. Adding to the generators with batteries group system, new auxiliary systems and inter-connecting loads, generators and SRE, avoid the power outage to be an accident initiator event.

1. INTRODUCTION

The IEA-R1 research reactor at the Institute of Energy and Nuclear Research (IPEN) is an open pool research reactor, built and designed by the US company "Babcock & Wilcox", having demineralized light water as coolant and moderator and beryllium and graphite, as reflectors.

The main purpose of the IEA-R1 reactor is the production of radioisotopes for use in medicine, nuclear materials and fuels testing, irradiation of samples with neutrons, conducting fundamental research in areas such as physics, radiochemistry, radiobiology, activation analysis, training of human resources at the level of graduation and training of skilled personnel for operation of the reactors.

The first criticality of the reactor took place in September, 1957. Originally, the reactor was designed for operation at 5 MW, but most operations occurred at 2 MW, until 1997. Between 1995 and 1997, the reactor received a series of modifications to appropriate it in terms of

security operation at 5 MW. In 1997, the reactor received a provisional authorization to operate a 5 MW.
The reactor pool is located within the reactor building. It has a total volume of 272 m^3 and is divided into two compartments: (a) operating compartment and (b) storage compartment of irradiated fuel elements. The two compartments can, optionally, be isolated with the use of a gate.
The IEA-R1 core reactor is, basically, constituted by a set of fuel elements, MTR type, which is submerged in the pool and suspended by a metal frame. The reactor is moderated and cooled with light water.
The reactor power control is performed by the movement of neutron absorbing elements (AE) within the reactor. Each element contains two absorber blades, coated by an Ag-In-Cd Ni alloy. The drive of absorbing elements for the reactor control is done by electromechanical mechanisms, independent from the absorbing element. These absorbing elements have, also, the function of ensuring the safe shutdown of the reactor when needed ("SCRAM"), with the free fall of all absorbing members through clearance electromagnets that support these elements. The reactor has only this means of reactivity control with two functions (control and security). An absorber element exercises the functions of control and security and the other three perform only security.

2. MATERIALS AND METHODS

The Electrical Energy System it consists of 4 (four) generators type "UPS", (diesel generator set), and Uninterruptible Power Module.

2.1 IEAR-1 Electric Energy System

Designed to meet the electricity demand required by the loads of the reactor (Security systems and systems not related to security) in the different situations the plant can meet, such as during startup, normal operation in power, shutdown, maintenance, exchange of fuel elements and accident situations. The system, also, feeds the Van de Graaff accelerator and nuclear physics experiments (experimental benches).
It is convenient to point out that the information contained here in is directed to display the functional adequacy of the system to the plant, as well as to indicate the degree of compliance of its designs with the rules applicable to industrial facilities, basically, rules established by the Brazilian Association of Technical Standards (ABNT).
The Electricity System is, normally, powered by the means of electricity (Eletropaulo), via Jaguaré substation. This extension is connected to IPEN underground distribution network, which is configured with two interlocking rings. In case of any electricity supply failure, the charges that should maintain the reactor operation continuity, including the experimental stands, are now powered by local sources of electrical power (diesel generator set, "UPS" set or Uninterruptible power module).
The Electricity System is divided into three (3) distribution systems. Each system takes into account the desired level of availability in the supply of electric power to the loads of the reactor, in order to maintain continuity of the same operation. These distribution systems are:

2.1.1 Normal Electric Distribution System (NEDS)

It can tolerate interruptions of a long period in the supply of electricity, without causing risks to the safety and continuity of operation. The NEDS is powered entirely by the mains: regular electricity supply.
Main charges:
- Air conditioning
- Holding tank Pumps
- Outdoor lightingogical

2.1.2 Essential Electric Distribution System (EEDS)

This system can tolerate short-term electricity supply interruptions, so it is essential to the security and continuity of the operation.
EEDS is fed by the regular network in the event of failure, providing the system is powered by conventional generators group.
Main charges:
220V: - Normal reactor building lighting
- Exhaust system and normal ventilation
- Overhead crane
440V: - Secondary pumps
- Fans of the towers
- Air compressor
- Swimming pool Isolation System Valves

2.1.3 Vital Electric Distribution System (VEDS)

Interruptions in the electrical power supply cannot be tolerated; therefore, this system is vital to the security and continuity of operations.
The VEDS is fed by the regular network: in case of failure occurrence, the system will be powered by generators of the group "UPS's" and Uninterruptible Power Modules.
Main charges:
220V: - Control Desk
- Firefighting system
- Table of nuclear physics
440V: - Primary Pumps
- Emergency Exhaust System
- The SRE Valves

2.1.4 Emergency Cooling System (SRE)

The Emergency Cooling System of the IEA-R1 (SRE) Reactor has the function of removing residual radioactive decay heat from the reactor core so as to avoid melting of the fuel elements, should a Refrigerant Loss Primary Accident (APRP) occur in the empty reactor pool and the core be totally or partially uncovered. It is considered as conservative basis for the SER project, for the fuel plate coating integrity, to keep a limit of 500 C temperature, in all fuel plate surfaces, has to be ensured.
The SRE consists of two elevated tanks, pipes, valves and a distributor with spray nozzles, in addition to the necessary instrumentation for measuring, actuation and monitoring, and a

distributor for performing periodic tests. A valve ball, with 03 (three) copies manually operated, is used to implement periodical tests with the distributor for testing. This valve has a key type micro-switch that sends signals to the scorer's table warning positioning status and preventing the reactor to be turned on if it is not properly positioned for cooling the core (core position). The SRE is designed to have automatic operation and passive action or manually by gravity through reservoirs located in a high position, in relation to the core, Figure 1.

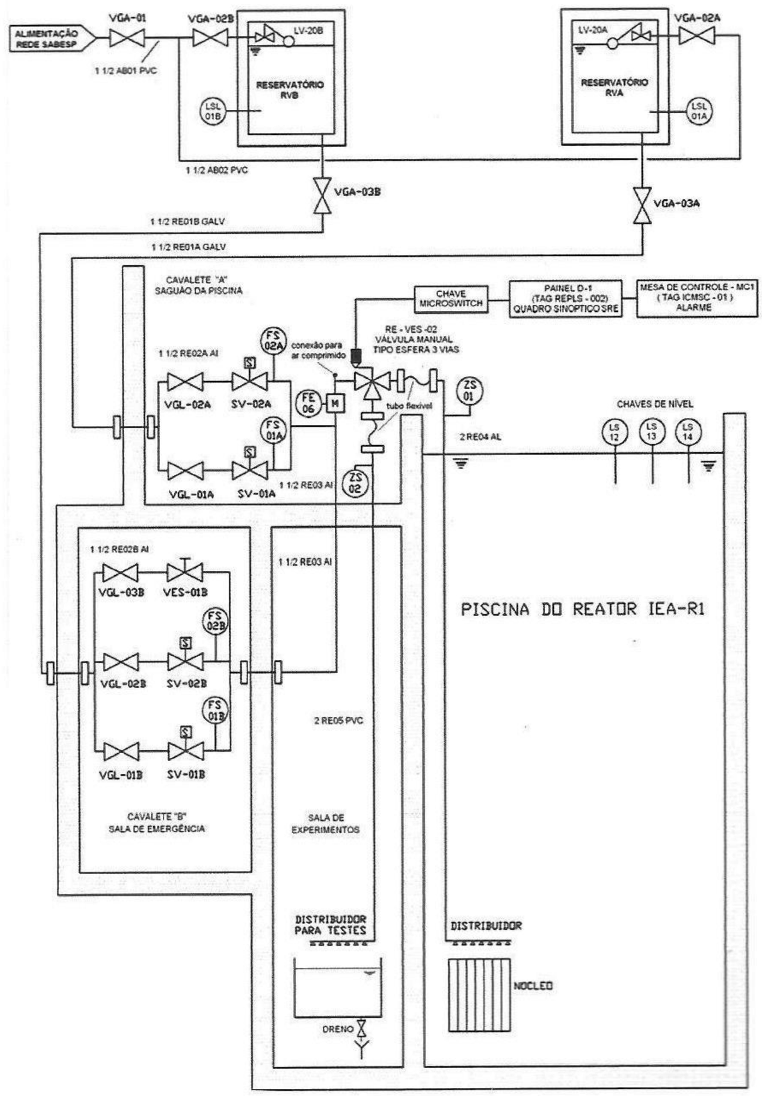

Figure 1: Emergency Cooling System

The RVA and RVB reservoirs are supplied with drinking water from the normal network water supply (SABESP). These reservoirs, separately, feed two pipe branches of 1 ½ "in diameter, which are sent in air path and follow, separately, to the building of the reactor, forming two independent companies. The branch A that does the feeding enters the reactor building directly at the pool hall, where the solenoid valve assembly 01A and SV-SV-02A are also located, with respective VGL-01A-VGL02A regulation valves. The branch B enters the reactor building through the emergency room, where the set of solenoid valves SV-01B, SV-02B, the manual ball valve VES-01B, the 01B VGL, VGL-02B-03B and VGL regulating balls are located. The branch A with the branch B lies within the pool hall, forming a single branch, containing the EF-flow meter 06, which runs along the edge of pool ending at 03 valve (three) RE-VES-02 pathways that directs the flow to the core or to carry out periodical tests. Flexible pipes connect this valve to the core water dispenser or water dispenser for testing. As mentioned above, SRE branches are independent supply lines and redundant. Each branch is supplied by a tank and has two valves of the solenoid type, also, redundant and branch B has, also, a manual ball valve. Each branch operates with only one of the open valves, meeting 100% of the water demand required by the system design conditions. Details of the project arrangement of the pipes are shown in the documents of Linic Engenharia Ltda [16].

The SRE has been designed to cool the core of the reactor IEA-R in the event of an accident with loss of coolant, after continuous operation for 120 hours at a power rated at 5 MW. The core has 20 Fuels Standard elements, 4 Fuel Control Elements and a central irradiator that in some Beryllium situations may be replaced by a standard fuel element. The core of the heat curve decay whose values were calculated using the ORIGEN code, also, shows another heat curve decay for values increased by 20%, which was used for the sizing of SRE.

In the system design, it is considered that the axial and radial heat flow profiles in the core of the reactor in off condition are the same as in normal operation, presented in [17]. It is, also, considered an uncertainty of + 10% at peak factors, calculated by multiplying the entire axial profile in the hot channel by a factor of 1.1.

Based on these conditions and in experiments reported in [18-20], it was settled one minimum flow rate of 3 m3 / hr (0.83 kg / s), for a minimum of 14 hours for the SRE. In accident analyses, it was established a time of 13.5 hours after the discovery of the core problem so that it can be cooled by natural air convection. The value of 3 m3 / h (0.83 kg / s) is set in VGL-03B regulating the installed upstream of the manual VES -01B ball valve. The system now operates with this flow after a time greater than 30 minutes from the start of the SRE activity, when solenoid valves are closed. The other regulating valves, VGL-01A, VGL-02A, VGL-01B and 02B-VGL, should be fully open, allowing a higher flow rate to 3.5m3 / hr (0.97 kg / s) if only one of the solenoid valves is actuated, and a top vent 4m3 / hr (1.11 kg / s) in the first operation period with the four solenoid valves open.

The flow rate of 3 m3 / h (0.83 kg / s) was optimized, experimentally, and considering surpluses for compensate distribution losses, the irrigation of the control elements and the effects of shading in the core, ensuring a higher minimum flow of 45 cm3 / min. (0.00075kg / s) per plate fuel. System sizing calculations are presented in reference [21] and the main results of experimental tests performed with a Fuel element model can be electrically heated plates, found in references [18] and [20].

2.3 Methods

It comprises methods and approaches used in the IEA-R1 reactor safety analysis. In general, this analysis has been a widely used approach, disseminated and accepted when considering possible initiating events of accidents: it applies a deterministic technique for assessing the consequences of such incidents. Thus, the methods used to identify and select primer accident events, methods of analysis of the accidents, including the sequence of events, transient analysis and radiological consequences are, here, described. Finally, the acceptance criteria for radiological doses will, also, be outlined.

2.3.1 Identification Methods and Key Initiator Events

The method used to identify postulated initiating events, i.e. the events which may lead to accident scenarios, are based on Safety Series 35 [1] and [2] and they consist of several steps, including the identification of a preliminary set of initiator events, the elimination of improper events, the classification of events into categories and the identification of limiting events (an event with a worse result than one category). Detailed studies of accident analysis will be, here, approached.

The preliminary list of accident initiating events was obtained from the generic list, for research reactors, listed in Table I of the Safety Series 35 [1] and [2] . From this list, inconsistent or inappropriate initiating events have been eliminated. They are: (a) the not credible (impossible to occur in the IEA-R1) accidents; (b) very rare events (initiating events whose occurrence frequency is so low that it may be rejected in probabilistic field) and (c) events that result from a combination of mutually independent events with low frequency of occurrence.

To this short list, new events from the 39-year-IEA-R1 reactor operation experience and the examination of safety reports were incorporated. Similar facilities, such as Democritos Greek Research Reactor [3] Union Carbide Research Reactor [4], Oak Ridge Research Reactor [5] Omega West Reactor [6] MIT and Research Reactor [7] have, also, been covered. After the list had been completed, the initiating events were grouped into categories that bring together events that exert similar influence on the reactor behavior and to which similar calculation models may be used. This list was selected for detailed analysis, the initiating events and limiting features suspected to belong to one or more categories, i.e., potential consequences superior to other accident initiating events of a same group.

2.3.2 Analysis Methods

The sequences of events were evaluated from the occurrence of the triggering event until the state damage end in the reactor, following the whole course of the accident.

2.3.2.1 Analysis of Event Sequences

The evaluations were considered both human interactions, as systems important to eliminate or mitigate the consequences of the accident, including the reactor protection system and safety systems. In addition, they considered the primary barrier, consisting of the cladding of the fuel, the pool of the reactor and the confinement.

During the study of event sequences, the following items were identified:

Significant occurrences over time, for example when shutting down the reactor, initial and final insertion of the safety bars, etc.: correct or incorrect operation of the reactor instrumentation and control in normal conditions; actions required from the operator; conditions for the completion of the analysis, including, for example, situations in which stable conditions are achieved (without exposure or release). Three major security functions were evaluated: reactor shutdown, fuel cooling and confinement of radionuclides, including an indication of the correct operation of the reactor protection system and safety systems, simultaneously, and their flaws.

The analysis was performed considering: effect of a single fault; Qualification System (or lack of qualifications) under accident conditions; protection and security systems, including reliability and quantitative parameters; support systems, such as electrical system service and emergency ; redundant shutdown parameters; actions of independent systems; operator actions (e.g response times, presenting information in a console, etc.).
The conventions were used to determine the sequences of events that are beyond design basis and are, therefore, excluded from the detailed analysis based on: Qualitative arguments justifying the exclusion of events whose occurrence is impossible or that are not credible for the reactor; qualification of the facility or the reactor systems to resist the effects of the event; probabilistic arguments.
The effects of human errors have, also, been considered and include: identifying their causes; evaluating their effect both as an initiator of an accident, and, also, as aggravation element of an accident in progress; critical evaluation of the assumptions adopted with respect to the response of the reactor systems during the accident sequence. The study of a limiting event sequence, i.e. the study of a sequence that has undergone detailed analysis, with use of transient analysis methods. The study was based on: conservative assumptions made in the classification of categories, such as non-give full credit to systems of mitigation actions or the operator's response; conservative assumptions to ensure that all sequences of a category were covered; in methods for limiting the choice of sequences for a group of events it is the entire class and, not only, a specific sequence, including those who have the most severe consequences.

2.3.3 Analysis of Transient

After defining the limiting events, a quantitative analysis of each of them was held. The definition of these events was based on a detailed study of the installation, similar studies performed in reactors and examples presented in the IAEA-TECDOC-643 [8].
For the analysis of transient and quantitative safety analysis methods, deterministically based, whenever possible, well known and spread computer programs were used. A very important factor for the transient analysis is the initial condition of the reactor in the start of the event. In setting the initial condition, conservative assumptions were adopted. Thus, the following conditions were used: expected core configuration with the least number of elements within a period of two years; for calculating the radionuclide inventory and power, decay was assumed with an average fuel burn of 30% in the core; the hot channel was modeled in terms of peaking factors and nuclear engineering; there was 10% overpower before the accident. For the analysis of the transients reactivity insertion and pump stop, the PARET code was used [9], since it is suitable for analysis of these types of events in reactors pool type.

2.3.4 Analysis of the Radiological Consequences

To evaluate the radiological consequences in the case of accidents involving core damage, conservative deterministic methods were used. The IAEA-TECDOC-643 [8] and analyzes for similar reactors have been used as reference models and parameters.
For the assessment of atmospheric dispersion and consequent radiological doses around the installation, the computer program will draw [10] implemented in the center was used in the Nuclear Engineering (CEN) of IPEN. The models used by this code are recommended by licensor organs and are based on the following tabs, Regulatory Guide 1.4 [11], Regulatory

Guide 1.145 [12] and NUREG / CR-3332 [13]. To evaluate the dose rates inside the reactor building and its surroundings, due to direct radiation, the same methodology at the Reactor Physics Group CEN/IPEN for shielding calculation of nuclear reactors was used. This methodology consists of a network coupled to nuclear codes (computer programs) that are used for the radiation transport solutions.

2.3.5 Radiological doses of Acceptance Criteria

As criteria for acceptance of radiological doses caused by accidents, the RESOLUTION CNEN-09/69 [14] was used, which dictates the Standards for Selection of Locations for Installation Power Reactors. In this resolution, exclusion zones and low population zones are defined.
The exclusion zone is the area belonging to the heritage of the carrier, surrounding the reactor, where the operator has the authority to determine all activities, including removing personnel. In the exclusion area, in case of an accident, the full dose of full body radiation cannot exceed 250 mSv and the inhalation radiation thyroid dose cannot exceed 3000 mSv for an individual located at a point on the outer boundary line, for two hours from the beginning of the maximum postulated accident.
The sparsely populated area is the sparsely populated area adjacent to the exclusion zone. In this area, the total number of people should allow the formation of protective measures to be taken in case of a serious accident. The limits of radiological doses to an individual in a low population zone are the same for the exclusion area. However, these doses must not be exceeded for an individual located on a point of its boundary line,through the cloud passage of time resulting from the release of radioactive radionuclides, due to the maximum postulated accident.
Maximum Postulate accident is the accident resulting from a possible sequence of events, the consequences of which will not be exceeded by any other possible accident, except those accidents whose probability of occurrence is so small that they can be considered impossible to occur.

2.4 Selection of Initiator Events

Listed below, the initiator postulated events for possible accidents in the IEA-R1 reactor. For each of the possible causes, triggering events are identified: the sequence of the incident and possible consequences are in chapter 16 Safety of the Analysis Report IEA-R1, butl only give a brief summary of the item with respect to loss power supply will be given It is expected that this list is complete concerning events primers. These events are divided into the following eight categories:
Loss of electrical power;
Excess reactivity insertion;
Loss of core flow;
Loss of primary coolant;
Wrong maneuver or equipment failure;
Special internal events;
External events;

Human error.

3. RESULTS AND DISCUSSIONS

The loss of normal electrical power to the IEA-R1 reactor is very common, as there is a single power line utility (Eletropaulo) to feed a single substation for the whole University City. IPEN is powered by a line which is independent from this substation. In the case of an Electric External Power Loss the IEA-R1 in the reactor building, there may be different sequences of events as described subsequently.

3.1 Failure of Normal Electric Distribution System

When the supply of external energy in the IEA-R1 building falls, the Vital Power Distribution System enters immediately, as described in Chapter 9. This system has a "UPS" type generator that will continue supplying power through batteries at the reactor control desk, and a "UPS" type generator, which has a flywheel coupled to an electric generator. This generator should keep the power electric circulation pump of the primary circuit, during the initial seconds of this event occurrence. The primary pump is kept in operation so that there is continuous cooling of the IEA-R1core. After about 10 seconds of the loss of normal electrical energy, it comes into operation the diesel-generator set belonging to the Vital Electrical Distribution System, which should, in turn, supply the electricity to the reactor control desk and the engine of the primary pump. In the sequence, the diesel group of generators of the Essential Electrical Distribution System enters in operation, supplying the other loads of the IEA-R1 building. After the electrical system operation sequence described above, the reactor will continue in normal operation, with the supply of electricity carried out by the Power Distribution Vital and Essential systems.

3.2 Failure of the Essential Electric Distribution System

This may occur when the external power supply of the Power Distribution Essentials system does not come into operation, as expected. In this case, the Vital Power Distribution System will continue supplying power to the reactor control table and circulation pumps of the primary circuit. In this case, there will be an immediate and automatic shutdown of the reactor for performance of the protection system. The reactor shutdown can be accomplished, in this case, by two ways. The first is the reactor shutdown, a few minutes after the event, by increasing the temperature of the pool water due to the loss of the cold source, i.e, secondary circuit loss. The other is a decision that may be taken by the responsible operation supervisor, who will be accompanying the behavior of the nuclear control desk and thermal-hydraulic parameters.

3.3 Failure of the Vital Electric Distribution System

In the case of loss of external power supply and the consequent failure of "UPS" generators and the Vital Electric Distribution System, it will occur, consequently, the automatic shutdown of the reactor. In case of failure of the "UPS" - 440 V generator set feeding the circulation pump, the primary circuit without failure in the "UPS" - 220 V generator set will provide electricity to the scorer's table and the reactor will be shut down for failure of

electricity supply to the primary pump. It was found, experimentally that the circulation pump of the primary circuit remains in operation, with decreasing rotation, for about 80 seconds after the start of power loss due to actions of the flywheel coupled to its shaft. However, although the flywheel acts for 80 seconds, it was also verified, experimentally, that the uncoupling of the convection valve occurs 24 seconds after the incident causing, at this moment, a reversal of the forced circulation flow downward for a natural movement upwards. It was observed that this experiment was carried out with an initial flow rate of 2,600gpm. With a flow rate of 3,000 gpm, the time involved will be slightly higher.
When there is loss of electrical supply to the primary pump the reactor shutdown signal is given. The maximum drop time of the control rods is about 1 second, from the shutdown signal.
If, after the loss of electric power supply, a failure occurs in "UPS" - 220 V generator set, which supplies electricity to the scorer's table, the reactor will be off, automatically, but it will continue to be cooled by the forced circulation fluid promoted by the primary pump. When falling the supply of external energy in the IEA-R1 the building falls, it enters, immediately, into operation the Vital Electrical Distribution System, consisting of 4 (four) "UPS" type generators that will continue supplying power to the reactor control table, the core cooling system and other security systems.

4. CONCLUSIONS

Studies have shown that, adding batteries to the system of generators, new auxiliary systems and inter-connecting loads and generators, the power outage will not be an accident initiator event.
By the late 80s, the reactor safety systems received as power supply only a power source: the control table was stocked with few vital 220V electrical systems, in the absence of electricity, powered by "UPS" 220V engine-generators.
In the years 1989 and 1990, the electricity system was updated and adapted to the power increase of the reactor reforms and, also, meeting the technical standards of the ABNT. This electrical interconnections reform was made in the various security systems making them have redundant power supply, taking the earlier example of control table, which besides being fueled by a vital electrical system. Failure to vital electric system is provided for essential electric system. Another example is the Emergency Cooling System (SRE).
In Fukushima, due to a large tsunami, normal electric power system was destroyed and the emergency generators did not come into operation causing the cooling system to shut down the reactors and causing a terrible accident. In the IEA-R1 reactor, this type of accident would not happen because the SRE action would avoid this type of accident.
SRE has the function of removing residual heat from the core to prevent the melting of fuel elements in the event of loss of refrigerant to the core. It consists of two elevated tanks with potable water that will work, automatically, when there is a lack of cooling for the reactor core, pouring water, continuously, on the core with a flow rate of 3m3 / h for fourteen (14) hours, long enough for the fuel elements cooling. Hence, with these upgrades the IEA-R1 reactor is safer and with longer lifetime.

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