ITALY

Topical Peer Review 2017

Ageing Management

National Assessment Report

2017

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This National Report has been prepared by the National Center for Nuclear Safety and Radiation Protection of ISPRA (Institute for Environmental Protection and Research), which carries out the functions of national competent regulatory Authority for nuclear safety and radiation protection

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00 Executive summary

As known, in Italy all NPPs and other nuclear installations were definitively shutdown many years ago and they are now at different stages of decommissioning. Only five research reactors remained into operation.

The Italian National Assessment Report for the Topical Peer Review 2017 on Ageing Management provides an overview on the national regulatory framework applicable to research reactors and describes the ageing management approach for two reactors, the TRIGA RC-1 reactor (1MW_{th}), operated by ENEA (Italian National Agency for New Technologies, Energy and Sustainable Economic Development), located in the Research Centre of Casaccia in Rome, and the TRIGA MARK II (0,250 MW_{th}), operated by the Applied Nuclear Energy Laboratory (LENA), University of Pavia. The TRIGA MARK II reactor has been selected on voluntary basis, being its power below the reference level of 1 MW_{th} established in the Technical Specifications as the threshold power value for a research reactor to be included in the TPR.

According to the legislative decree n. 137/2017, with which the EU Directive 2014/87/Euratom has been transposed into the national legislation, the NAR has been prepared by the National Center for Nuclear Safety and Radiation Protection of ISPRA (Institute for Environmental Protection and Research), which carries out the functions of national competent regulatory Authority for nuclear safety and radiation protection.

Legislative Decree n° 45/2014 establishes a new competent regulatory Authority in the field of nuclear safety and radiation protection, that is the National Inspectorate for Nuclear Safety and Radiation Protection (ISIN); according to the same Decree, until the establishment process of ISIN will be completed, the functions of competent regulatory authority continue to be carried out by ISPRA, through the National Centre for Nuclear Safety and Radiation Protection.

The structure of the report follows the Technical Specifications of the WENRA RHWG for the National Assessment Reports on Ageing Management dated 21 December 2016, and has been prepared taking into account those parts considered applicable to research reactors. In particular, the following specific topics have been included: Electrical cables, Concealed pipework, Reactor pressure vessel, Concrete containment structures, with some adaptations applicable to the case of research reactors.

The National Report has been prepared on the basis of specific assessment reports prepared by the Licensees on specific request of ISPRA.

The present Italian legislative and regulatory framework related to nuclear and radiation safety is an evolution of rules and standards that begun in the early 60ties. It applies to all nuclear installations and activities, including research reactors. The system benefits of the long experience of licensing and operation of NPPs and other installations as well as the transposition of EU Directives. Main principles reflected in the legislative and regulatory framework are assignment of primary responsibility for safety to the licensee, independence of the regulatory function, a system of licensing and authorizations, a system of regulatory supervision and inspection, a system of enforcement accompanied by penal and administrative sanctions in case of non compliance with legislative provisions and license conditions.

In this regard an international peer review implemented with an IAEA IRRS mission took place in 2016.

For research reactors conditions attached to the licence request the licensees to prepare every five years a report on the status of conservation of the installation and on its operating experience. In 2013 ISPRA has requested the licensees to improve these reports in line with periodic safety review requirements, as established in the Code of Conduct for the safety of Research Reactors.

Taking into account the age of the reactors particular attention is devoted from the regulatory point of view to the proper management by the licensees of aging mechanism, either related to physical ageing, leading to the degradation of the physical characteristics of SSCs, or to non-physical ageing, related to technological obsolescence.

Aging management has been implemented along the years as result of the application of the surveillance rules associated to conditions and technical specifications attached to the licence. This has lead to the implementation of several upgrading measures.

As far as concern TRIGA Mark II, it is recognized that in the recent times an Integrated Management System is implemented in accordance with International Standard ISO 9001. On that basis an ageing management programme (AMP) is also in place. The outcome of the comparison of the IAEA guideline SSG-10 *"Ageing Management for Research Reactors*" with the IMS in place leaded to the drafting of a first a road map for all the activities not already included or providing adjustments to incorporate ageing in the main management system.

The proper application of this AMP is monitored by ISPRA.

With regard to TRIGA RC-1, ISPRA has taken note of the Licensee plan to implement in 2018 some extraordinary maintenance and upgrading modifications on infrastructures of the reactor building, on the electrical power supply and on the instrumentation and control system. For this purpose the reactor will be put in temporary shutdown conditions. An Ageing Management Programme to be applied in the future, after the above mentioned interventions will be completed, has been prepared.

Application documents related to the licensing of the above activities will be submitted soon.

ISPRA has taken the decision to request the Licensee the conduct of the first inspection activities identified in the Ageing Management Programme for Electrical Cables, Hydraulic Circuit and Container during the envisaged shutdown phase.

00.1 Preamble

The present National Assessment Report for the Topical Peer Review on Ageing Management has been prepared by the National Centre for Nuclear Safety and Radiation Protection of ISPRA, from now on ISPRA, which carries out the functions of national competent regulatory Authority for nuclear safety and radiation protection. The report has been prepared in relation to two research reactors currently into operation in Italy, the TRIGA RC-1 reactor (1MW_{th}), operated by ENEA (Italian National Agency for New Technologies, Energy and Sustainable Economic Development), located in the Research Centre of Casaccia in Rome, and the TRIGA MARK II (0,250 MW_{th}), operated by the Applied Nuclear Energy Laboratory (LENA), University of Pavia. The TRIGA MARK II reactor has been selected on voluntary basis, being its power below the reference level of 1 MW_{th} established in the Technical Specifications as the threshold power value for a research reactor to be included in the TPR.

It is based on the specific documents prepared by the Licensees upon request of ISPRA and on the results of ISPRA assessment and oversight activities.

The report has been developed with reference to the Technical Specifications on National Assessment Reports according to a graded approach, taking into account the specificities of research reactors. In particular, the following chapters has been included: 03 – Electrical cables, 04 – Concealed pipework, 05 – Reactor pressure vessel, 07 – Concrete containment structures, with some adaptations applicable to the case of research reactors.

01
General information

01.1 Nuclear installations identification

The two following tables provide an overview of research reactors present in Italy. In Table 1 a list of research reactors that have been decommissioned or are in the process of being decommissioned or in permanent shutdown. Tables 2 reports a list of research reactors that are into operation.

Table 1- RRs decommissioned, in decommissioning or in permanent shutdown

Facility Name	Thermal Power (kW)	Type	Status
RITMO REACTOR (RC-4)	0.01	POOL	Decommissioned
ROSPO 2	0.20	POOL	Decommissioned
Subcritica Structure	0.00	SUBCRIT	Decommissioned
RANA	10.00	POOL	Decommissioned
RB-1	20.00	CRIT GRAPHITE	Decommissioned
RB-2	10.00	ARGONAUT	Decommissioned
AVOGADRO RS-1	5,000.00	POOL, MTR	Decommissioned
RB-3	0.10	Zero Power (DO_2)	In decommissioning[1]
ESSOR (CP-5)	25,000.00	HEAVY WATER	Permanent Shutdown[2]
ISPRA-1	5,000.00	HEAVY WATER	In decommissioning
GALILEO GALILEI RTS-1[3]	5,000.00	POOL	In decommissioning

1 Decommissioning operations have been completed. Site release still to be accomplished.
2 Fuel has been removed from reactor
3 This Research Reactor is managed by the Administration of defence

Table 2 - RRs in operation

Facility Name	Power (kW)	Type	Status
AGN 201 COSTANZA	0.02	HOMOG (S)	Operational
LENA, TRIGA II PAVIA	250.00	TRIGA MARK II	Operational
RSV TAPIRO	5.00	FAST SOURCE	Operational
SM-1	0.00	SUBCRITICAL	Operational
TRIGA RC-1	1,000.00	TRIGA MARK II	Operational

Among the operating research reactors, the only one under the scope of the National Assessment Report (NAR) is the TRIGA RC-1 reactor operated by ENEA .

The TRIGA Mark II reactor, operated by the LENA of the University of Pavia, that is below the power level of 1 MW_{th} assumed as reference for inclusion in the TPR, is addressed on a voluntary basis.

ENEA – TRIGA RC-1

Since June 1960 a TRIGA (Training Research Isotopes General Atomics) Mark II reactor, built by General Atomics, is in operation at the ENEA Casaccia Research Center that is located in S. Maria di Galeria, a small area in the municipality of Rome.

This reactor, named RC-1 (Casaccia Reactor 1), was operated at 100 kW power up to August 1965. At this power the maximum flow in the core was 4 10^{12} n/cm^2 s.

During the period from June 1960 until August 1965 the TRIGA RC-1 reactor was used for research in the field of Physics of Solids and Nuclear Physics, as well as for the preparation of radioisotopes used for physics, biology and chemistry experiments.

In the summer of 1965, work began to increase the reactor power to 1 MW.

The first criticality in the current operating configuration of 1 MW was reached in July 1967.

Reactor Operation

From the date of the first criticality in the current 1 MW_{th} operating framework (July 1967), until the date of issue of the operating license (Ministerial Decree No. VIII/27 of September 10^{th},1970), the Safety and Control Division of CNEN (the national competent regulatory authority for nuclear safety and radiation protection at that time) authorized the provisional operation of the reactor for performing nuclear power tests (measurements of the temperature prompt coefficient, determination of spatial distribution of the neutron flow, measurement of power effects, detection of the radiological parameters, etc.) and for the functional verification of experimental equipment coupled directly or indirectly to the core.

In the current year (2017), the TRIGA RC-1 reactor was operated for a total of a few MWd due to planned replacement and maintenance activities of nuclear instrumentation for reactor control and monitoring. These activities include a series of interventions aimed at adapting the SSCs of the TRIGA RC-1 reactor due to ageing. At present TRIGA RC-1 is in a test phase to check the electronic components installed to replace the old ones.

In the coming months some minor operation activities will be carried out aimed to:

- verify the effective capability of the production of Technetium-99m by means of Molybdenum targets irradiation;
- evaluate electronics components radiation damage in the frame of a cooperation with Italian Space Agency (ASI).

After the completion of above mentioned activities, in the second half of 2018, the reactor will be put in temporary shutdown condition in order to implement some interventions based on ageing-related evaluations and involving both the structural part of the reactor building and the electrical power supply part, in addition to the aforementioned modernization of the instrumentation and control system.

In detail, the foreseen activities are the following:

a) **Seismic resistance verification:** the aim is to verify the maintenance of the seismic design characteristics of the reactor building as well as the relevance of some small cracks on the external surfaces of the building itself;

b) **Infrastructures-related activities:** analyzing the outcome of the activities performed in the previous phase, the maintenance interventions will be designed and implemented on the reactor building;

c) **Fire protection compartments:** in order to obtain the fire prevention certification according to national legislation, the compartmentalization of some rooms and the renewal of electrical panels was requested. The activities will be carried out in parallel with the interventions on the reactor building;

d) **Control room console renewal:** the complete renewal of the instrumentation was initiated some years ago following the “step by step” approach: in this way the work procedures are simplified and both costs and facility unavailability are reduced. At the moment many parts of the facility have been replaced with new equipment (i.e. radiation and contamination monitoring systems, temperatures monitoring and storage, water flow measurement systems, log amplifier and rate meter). Next steps are the replacement of the two independent linear amplifiers and the safety amplifier . In the next months the procedure to procure secondary accessories of the main console will start: relé actuators racks, alarms racks, console internal connection cables, control rods magnets and driving motors power supply and so on.

The licensing application for this upgrading programme is under preparation.

LENA TRIGA MARK II

The pulsed TRIGA MARK II reactor is installed at the University of Pavia, Applied Nuclear Power Laboratory (LENA), and reached the first criticality on November 15, 1965; the operation of the facility at present is authorized with the Decree of the Ministry of Economic Development n.VII-285 of May 22, 1990.

Licence history

1965 – Reactor goes operational

1970 – New licence and operational limits and controls

1976 – Licence extension

1983 – Licence renewal with re-evaluation of:

- plant response to design events
- environmental impact in standard operation conditions and in emergency
- technical requirements emergency plan
- evaluation of earthquake events
- evaluation of loss of power supply event
- evaluation of fuel element movement events
- revision of ventilation system
- revision of fire protection system
- revision of power supply systems
- revision of ICS's
- revision of waste management
- revision of radioprotection surveillance
- revision of operational documentation
- revision of the technical plant documentation
- revision of Operating Limits and Conditions *(*OLC's)

1990 – New licence

2005– New OLC's for the solid waste management

2013 – Code of Conduct implementation

The TRIGA MARK II reactor is in operation since November 1970; it is a pool type reactor with a core placed in a well filled with demineralised water surrounded by a graphite reflector. The core consists of a fuel-moderator element made up of an homogeneous mixture of uranium and zirconium hydrate, moderated and cooled by the well water. The fundamental feature of these fuel elements is their high negative temperature coefficient that automatically limits power in the event of power hikes. It is also controlled by 3 neutron absorbing bars. The reactor can operate at steady-state thermal power level up to 250 kW. At present, fuel elements are made of aluminium and/or stainless steel coated aluminium uranium and zirconium alloy.

The fuel is made up of a uranium alloy and zirconium hydroxide having 8% by weight of uranium enriched to 20% in U235. The atomic ratio H/Zr is 1.

Each fuel element contains 36.5 gr of U235 on average.

Burning poison is samarium oxide in the form of circular plates between two aluminium plates and is incorporated into the lower and upper part of each fuel rod.

The moderator is both zirconium hydride of the uranium alloy - zirconium hydride and water.

The reflector material consists of graphite:

- radial thickness 30.5 cm
- axial thickness 10.2 cm

In the central area of the core a water canal of about 5 dm is envisaged.

Three control bars are installed and operate all during the reactor operation; the excess reactivity does not exceed half of the total reactivity controlled by the bars.

Under cold and clean kernel conditions, a shutdown margin (a few hundred p.c.m.) is also ensured even with the most reactive bar fully extracted.

Fast reactor shutdown is for:

- power greater than 110% of the full scale of the linear channel;
- power greater than 110% of the full scale of the safety channel;
- an adjustable too short period up to a minimum of 7 seconds;
- Lack of power to control rooms.

There is an interlock that prevents lifting of control bars by counting on the pulse channel below 3 cps.

01.2 Process to develop the national assessment report

For the preparation of this report, ISPRA has requested the licensee of the research reactors identified for the TPR to submit a report providing specific information and data on individual facilities in compliance with the format of the TPR report, for parts applicable to research reactors.

Compared to the standard NAR index, since the two TRIGAs operating in Italy do not have a pressure vessel and containment structures, the contents of the relevant chapters have been adapted to the specific configuration of the two installations.

On the basis of the documents provided by the licensees and taking into account the results of its surveillance activities, the specific assessment conducted by ISPRA is also reported.

02
Overall Ageing Management Programme requirements and implementation

02.1 National regulatory framework

The present Italian legislative and regulatory framework related to nuclear and radiation safety is an evolution of rules and standards that begun in the early 60's. It applies to all nuclear installations and activities, including research reactors. The system benefits of the long experience of licensing and operation of NPPs and other installations as well as the transposition of EU Directives. Main principles reflected in the legislative and regulatory framework are:

- Assignment of primary responsibility for safety to the Licensee,
- System of licensing and authorizations,
- Comprehensive licensing procedure for decommissioning,
- System of regulatory supervision and inspection,
- System of enforcement accompanied by penal and administrative sanctions in case of non compliance with Legislative Provisions and License Conditions,
- Independence of the regulatory function.

The relevant part of current Italian legislative and regulatory framework, which is applicable to the construction, commissioning, operation and decommissioning for research reactors is constituted by:

- Act No. 1860/1962 on the Peaceful Use of Nuclear Energy, which establishes that the operation of nuclear installations, including research reactors, is authorized by the Ministry of Economic Development;
- Legislative Decree No. 230/1995, that is the main piece of legislation (with penal provisions) regulating in particular the licensing process and laying down radiation protection requirements for workers and the public;
- Presidential Decree No. 1450/1979, which contains requirements and procedure for the recognition of the capability for the technical conduction of nuclear installations.
- The Legislative Decree March, 4 2014, n. 45, implementing the Directive 2011/70/ EURATOM for the responsible and safe management of spent nuclear fuel and radioactive waste, establishes also the new competent Regulatory Authority in the field of nuclear safety and radiation protection, named National Inspectorate for Nuclear Safety and Radiation Protection (ISIN), to be set up with structures from the National Centre for Nuclear Safety and Radiation Protection of ISPRA, which continues to carry out the functions of competent regulatory authority until the establishment process of ISIN will be completed.

The Legislative Decree n. 230 of 1995, and the subsequent modifications (the last by *D.Lgs. n. 137 on September 15, 2017,* to transpose the EU Directive 2014/87/EURATOM establishing a Community framework for the nuclear safety of nuclear installations), is main piece of the Italian legislation on licensing process on nuclear installations.

The Italian regulatory body is a system made of the Ministry of Economic Development, which issues authorizations, and the National Centre for Nuclear Safety and Radiation Protection of ISPRA which is the national authority for technical regulation, independent review, assessment control and supervision in the field of nuclear safety and radiation protection of nuclear installations, including research reactors.

The Minister of Economic Development issues authorizations for nuclear installations and major practices involving the use of ionizing radiations; the operation licence and the decommissioning authorization are granted on the basis of conditions and technical specifications defined by ISPRA, taking into account the advises formulated by other Ministries and Authorities of the involved Region.

ISPRA (Institute for Environmental Protection and Research) is the competent regulatory authority for technical regulation, control and supervision in the field of nuclear safety and radiation protection of nuclear installations, provides support to civil protection authorities for emergency preparedness and response, ensures the fulfilment of the obligations arising from international treaties and conventions. ISPRA is entitled to issue approvals (e.g. of Detailed Projects and Plans of Operation) established in the more general framework of granted authorizations (operation licence, authorization of modifications, authorization of decommissioning operations).

Legislative Decree n° 45/2014 establishes a new competent regulatory Authority in the field of nuclear safety and radiation protection, that is the National Inspectorate for Nuclear Safety and Radiation Protection (ISIN); the Inspectorate will have strengthened authority, resources and independence; according to the same Decree, until the establishment process of ISIN will be completed, the functions of competent regulatory authority continue to be carried out by ISPRA, through the National Centre for Nuclear Safety and Radiation Protection.

Compliance with the legal requirements regarding nuclear safety and radiation protection is verified and enforced by the competent regulatory authority, currently the National Centre for Nuclear Safety and Radiation protection of ISPRA. The compliance is verified by reviewing safety analysis reports during the licensing steps and by supervising construction and operation, particularly through inspections.

Some definitions of the law related to research reactors are the following:

a) <u>nuclear reactor:</u> each apparatus intended for peaceful uses designed or used to produce a nuclear chain reaction, self-sustaining under normal conditions, even in the absence of neutron sources;

b) subcritical nuclear complex: each apparatus designed or used to produce a nuclear chain reaction, incapable of self-sustaining in the absence of neutron sources, in normal conditions or accidental;

c) research nuclear plant: every system equipped with a nuclear reactor in which the energy or fissile materials produced are not used for industrial purposes;

Research reactors.

1. For systems with a research reactor power not exceeding 100 kW thermal does not apply the procedure laid down in Articles 38 (Technical Examination) and 39 (Consultation with the Concerned Authorities). According to a graded approach a simplified procedure is foreseen for RRs with power below 100 kW_{th}
2. For research reactors of higher power the provisions related to all other nucler installations fully apply (Chapter VII of Legislative Decree n. 230/1995).

In general terms, the construction, commissioning and operation for nuclear facilities, including research reactors, is authorized by the Ministry of Economic Development with the involvement of other relevant ministries (Ministries of Environment, Interior, Labour, Health). The binding technical advice of ISPRA is requested by law. As said, at present in Italy there are only research reactors into operation or in decommissioning. The construction of new NPPs and research reactors is not envisaged.

The licensing procedure, laid down in articles 38 (Technical Examination) and 39 (Consultation with the Concerned Authorities) of Legislative Decree No. 230/1995, also related to research reactors involves the following steps:

- the applicant has to submit the documentation addressed to demonstrate that nuclear safety and health protection requirements are met: the requested documents are:
    - preliminary plans of the installation, complete with a topographical map, explanatory diagrams, drawings and descriptions of the installation and a preliminary study concerning the disposal of radioactive waste;
    - preliminary safety report, indicating the planned safety and protection measures;
- on the basis of the requested documentation, and other supplementary documentation it deems necessary for the purposes of the examination, ISPRA carries out its review and assessment of the preliminary safety report and of the preliminary study on the disposal of radioactive waste and prepares a technical report with the results of its critical examination;
- all the concerned Administrations shall communicate to ISPRA their opinions on the Preliminary SAR and the siting of the installation;
- taking into account any observations of the relevant Ministries (Ministries of Environment, Interior, Labour, Health), ISPRA expresses a final technical advice, specifying any conditions

and technical specification to be established for the execution of the project, and forward it to the Ministry of Economic Development which will issue the construction permit;

- ISPRA identifies those systems and parts of the installation considered relevant from nuclear safety and radiation protection point of view. For these systems and parts specific detailed projects have to be submitted to ISPRA for approval;
- construction is carried out under the technical supervision of ISPRA that verifies the compliance with the requirements of the submitted safety reports and the conditions issued by ISPRA;
- commissioning is executed in two logical sequential phases:
    - cold (non nuclear) tests (art. 43 of Legislative Decree No. 230/1995). Upon completion of all tests, a report of the tests shall be submitted to ISPRA that issues an appropriate certification declaring the positive results of the tests;
    - nuclear tests (art. 44 of Legislative Decree No. 230/1995). ISPRA issues the Certification of the outcome of the nuclear tests;
- the licensee updates the Preliminary Safety Report issuing the Final Safety Report;
- ISPRA transmits to the Ministry of Economic Development its final technical advice based on the commissioning program and the evaluation of the Final Safety Report;
- based upon the technical input, the operating license is issued by a Ministry's of Economic Development Decree. The operating license contains the management and technical specifications established by ISPRA.

For systems with a research reactor power not exceeding 100 kW thermal the procedure laid down in articles 38 (Technical Examination) and 39 (Consultation with the Concerned Authorities) does not apply, but a simplified one is foreseen.

With regard to facility modifications, art. 6 of Act No. 1860/1962 requires that any modification must be authorized. For modification it is intended any change in the characteristics of the nuclear facility compared to those resulting from Final Safety Analysis Report**.** The authorization is granted by the Ministry of Economic Development based upon the binding technical advice of ISPRA.

The process for modifications, when relevant for safety, is regulated by a specific Technical guide issued by ISPRA (T.G. No. 2). The licensee shall submit the documentation related to the modification, which will include its objective, the corresponding safety analysis, its impact on the safety of the nuclear facility and changes of the previously performed safety analysis. For relevant modifications the related detailed project has to be approved by ISPRA.

In relation to the personnel to conduct the nuclear facilities operation the Italian legislation states that it must follow an appropriate training program and its capacity to operate in a nuclear installation must be certified. The Operating Rules (Regolamento di esercizio), which defines positions in the licensee organization relevant to nuclear safety and radiation protection, are approved by ISPRA according to art. 46 of Legislative Decree No. 230/1995.

In addition, to certificate the licensee personnel qualification several examinations must be passed by each person. In particular, according to the Presidential Decree No. 1450/1979, three levels are foreseen to certificate the adequacy to be licensee, supervisor or technical director of a nuclear facility. The certification is issued by the competent local office of the Ministry of Labour after the verification of the technical and health skills as required by the law, performed by two specific commissions, respectively for medical and technical aspects.

For nuclear facilities, art. 49 of Legislative Decree No. 230/1995 requires a Plant Safety Committee to be established, whose composition has to be approved by ISPRA.

The Committee is composed of at least four members chosen among the technicians that oversee essential services for the operation of the plant.

The Committee has an advisory role, with the following tasks:

a) express prior opinion on any proposed modification to the plant or to its parts;
b) express prior opinion on any proposed change to the procedures of the facility;
c) express prior opinion on programs of experience, trial and extraordinary operations to be performed on the plant;
d) periodically review the performance operation of the plant, expressing its opinion together with any recommendations relating to safety and security;
e) developing the site emergency plan and provide at its any subsequent changes, in agreement with the provincial command of fire-fighters;
f) assist the Director on duty or the Plant Head in the adoption of measures that are necessary to deal with any event or abnormalities that may cause concern on the occurrence of a hazard to public safety or property damage.

The Operating rules and the Plant Safety Committee remain in place also during decommissioning activities.

For research reactors the Operating Licence, issued by the Ministry of Economic Development, establishes the condition that every five years a detailed report on the status of conservation and on the operating experience has to be transmitted to the Ministry of Economic Development and to the competent regulatory Authority.

With regard to the national institutional and regulatory framework, as well as to the regulatory body, an international peer review implemented with an IAEA IRRS mission took place in 2016.

02.2 International standards

ISPRA has requested licensees to submit an updated evaluation of safety which takes into account the recommendations of the "*Code of Conduct on the Safety of Research Reactors"*.

In this regard, the implementation of a periodic safety review, including ageing verification aspects, has been requested to be provided in the context of the five years report on status of conservation, whose preparation is made mandatory by the licence.

For the periodic safety review reference is made to the IAEA *"Periodic Safety Review for NPPs"* Specific Safety Guide No. SSG-25. This guide is applied according to a graded approach taking into account the IAEA *"Use of a graded approach in the application of the safety requirements for research Reactors"* Specific Safety Guide No. SSG-22.

In relation to ageing, the IAEA *"Ageing Management for Research Reactors"* Specific Safety Guide No. SSG-10.

02.3 Description of the overall ageing management programme

02.3.1 Scope of the overall AMP

WENRA I1.1 The operating organisation shall have an Ageing Management Programme (AMP) to identify all ageing mechanisms relevant to structures, systems and components (SSCs) important to safety, determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.

TRIGA RC-1 ageing management program

TRIGA RC-1 reactor has undergone various interventions to ensure its safe operation in compliance with the operating Licence technical specifications. A process devoted to identify and replace (or to upgrade) the equipment for monitoring and controlling the reactor operation has always been in place. In this process ageing effects and obsolescence of the technology have been taken into account. The results of the actions taken are detailed in the document *"State of conservation and operating experience of the reactor"*, recently integrated with the updated safety evaluation of the reactor (Periodic Safety Review).

The management of SSCs is based on the fulfilment of technical specifications established in the licence as well as on surveillance and good practice standards. The type of intervention implemented for each type of SSC depends on its classification from the safety point of view, according to the required reliability level.

Based on the technical specifications and on the application of the relative surveillance rules, the replacement of a component is implemented mainly if the results of a measure related to its operation significantly deviates from the nominal value, or from the range of values prescribed for that parameter, or if a failure is detected.

In the current reactor management system it is possible to outline some of the features of an AMP.

It is based on the identification of:

1. a precise hierarchy in management responsibility;
2. classification criteria for SSC relevant for safety;
3. methodologies for evaluating SSC status;
4. processes to ensure an increasing level of Quality Control.

The Reactor is managed by the Responsible Director, legally appointed by the Licensee. The Responsible Director is supported by all the management figures described in the Operating Rules, particularly by those that play a primary role in systems operation, monitoring and replacement in case of needs.

The second and third point of the list refer to the safety classification and the methodology for the verification of efficiency and operability of the SSCs according to their relevance in ensuring nuclear safety and radiation protection.

Three main classes can be defined:

- (essential) SSC for safety and protection system;
- (important) SSC in support of the safety and protection system;
- (not important) SSC not in support of the safety and protection system.

Each class, which groups several elements of the SSCs, is associated with a detailed control program, reflected in the technical specifications and in the surveillance rules. A key role for the periodicity is played by the control tests are performed:

- at the beginning and at the end of each working day;
- before each power ramp.

The first list of controls, carried out on a daily basis, includes both verification of the correct operation of the inspected system and the verification of the characteristic values of the parameters:

- cooling circuit:
  - flow,
  - conductivity,
  - contamination.
- air circulation circuit:

  - environmental X monitoring;
  - contamination from gaseous fluids monitoring;
  - corpuscular contamination β monitoring;
- visual checks on radioactive liquid discharge system;
- visual check on the reactor block;
- demineralised water generation system check.

Checks on the correct operation of the control system, of the intervention logics and reactor protection are performed following the second list, mandatory before any intervention on the control structures. All related alarms and fail-safe logics of the protection and control systems of the reactor are checked too:

- nuclear channels;
- SCRAM, rods return in position, threshold alarms retrieval, irregular high voltage and cables disconnection;
- many alarms regarding:
  - irregular low voltage of console instruments;
  - high level of environmental and/or cooling circuit radioactivity;
  - irregular water level in well and experience pool;
  - irregular water level in drainage well for radioactive discharge;
  - irregular water level in hot fuel repository;
  - irregular water level drainage pumps room;
  - irregular temperature of cooling fluid;
  - irregular flow of cooling fluid.

The outcomes of both check-lists must guarantee the correct operation, with values in the acceptance range, for all measured parameters. It's forbidden to operate the reactor if any alarm is present.

In addition to what described above, the requirements and the related surveillance rules, with a periodicity of at least one month or more, must be added. The planned periodicity of the surveillance tests (depending on the safety classification of SSCs) is reported in the following table.

Table 3 – TRIGA RC-1 - Parameters periodical check of SSC

SSC	Parameter	Periodicity
Control rods	Fall time	quarterly
	Interlocks	quarterly
	Scale change	quarterly
Nuclear channels	Calibration	quarterly
Dust filtering system	Efficiency	annual
Environmental monitoring	Functional test	monthly
	Calibration	half yearly
Gas and particulate monitoring	Functional test	monthly
	Calibration	half yearly

The improvement of systems supporting plant operators and, mainly, the replacement of analog instrumentation with instrumentation capable of storing data, provide an important support to the management of the reactor.

In support of the plant personnel responsible for verifying and deciding to replace any fault component or showing abnormal deviations from normal operation there are:

- the maintenance record, with historical data for the interventions performed for each individual component;
- the updated Periodic Safety Review.

The technical support to the operators is given by:

- contractors specialized in mechanical and electronic interventions;
- a workshop inside the plant structure with suitable instrumentation.

It can be stated that the effectiveness of interventions is actually the result of the synergy of several processes, more precisely:

- SSC status evaluation methods;
- documentary system for management of the interventions;
- SSC data acquisition system;
- organized intervention for anomalies and/or improvements.

As already mentioned, a plan of checks aimed to detect aging phenomena of the SSCs of the TRIGA RC-1 reactor is underway and will be added to the procedures and standards previously described.

The choice of the components to be addressed and of the methodologies to be used was made following IAEA Guide SSG-10 and applying the "graded approach" criterion. The following table describes the components, the inspection methodology and the periodicity supposed for TRIGA RC-1 to implement the ageing management program.

Table 4 –TRIGA RC-1 - Parameters periodical check of SSC

Component	Inspection Type	Ageing mechanism[1]	Replacement criticality
Vessel	Visual	1, 2, 4, 5, 6	A
Structure of core	Visual	1, 4, 5, 6, 7	A
Reflector	Visual	1, 4, 5	A
Control rods mechanism	Functional	1, 4, 5	C
	Instrumental		
Liner	Visual	1, 3, 5	A
	Instrumental		
Pool F.E. racks	Visual	1, 5	C
Primary circuit	Visual	1, 3, 4, 5, 6, 7	B
	Instrumental		
Purifying circuit	Visual	1, 2, 5	B
	Instrumental		
Secondary circuit	Visual	4, 5, 6, 7	B
	Instrumental		
Emergency Circuit	Visual	3, 4, 5, 6	A
	Instrumental		
Reactor building	Visual	2, 3, 4, 5	A
	Instrumental		
Biologic shielding	Visual	1, 2, 3, 4, 5	A
	Instrumental		
Ventilation	Functional	2, 5, 6	A
	Instrumental		
Experimental channels	Visual	1, 2, 4, 5	A
	Instrumental		
Reactor shutdown system	Functional	4, 5	C
	Instrumental		
Reactor protection system	Functional	4, 5	C
	Instrumental		

[1]The meaning of the values for the Ageing Mechanisms and Replacement Criteria fields is described in the Annex 1

Component	Inspection Type	Ageing mechanism[1]	Replacement criticality
Control system	Functional	2, 4, 6	C
	Instrumental		
Console	Functional	2, 4, 6	C
	Instrumental		
Radiation monitoring	Functional	2, 4, 6	C
	Instrumental		
Announcement system	Functional		C
	Instrumental		
Data acquisition system	Functional	4, 5	C
	Instrumental		
Power supply	Functional	5, 6	C
	Instrumental		
Emergency power supply	Functional	5,6	C
	Instrumental		
Handling elements tools	Functional	5	C
	Instrumental		
Fire system	Functional	5	C
	Instrumental		
Facility documentation		10	C

TRIGA Mark II - LENA Integrated Management System

In 2010, in order to continuously improve the quality of the reactor management and for the accomplishment of the stakeholder requirements (both under legal and commercial framework), LENA implemented an Integrated Management System in accordance with International Standard ISO 9001. This choice allowed to satisfy both national and international requirements (i.e. safe reactor operation and maintenance) and typical ISO 9001 requirements (as e.g. continuous improvement, users/stakeholders care and satisfaction) as well as complying with the IAEA GSR-3 "*The management system for facilities and activities*". Requirements are set out in an the integrated management system (IMS) and are binding on all levels of management personnel and staff within LENA. Amongst others, effective maintenance, ageing management and in-service inspection are some of the most important activities to achieve safety and maintain the intent of the design objectives during the operation of the facility. The experience in the application of IMS with regard to maintenance and ageing issues resulted particularly helpful, allowing a general improvement of the reactor operation and maintenance. As far as ageing is concerned, the introduction of an IMS leads to the following main benefits:

- better and more detailed in-service control plans were defined to examine plant components and systems for possible deterioration in their integrity to assess the safety margins and their acceptability for continued operation of the plant and to take corrective measures as necessary. The main systems and equipment important to safety of the plant are identified in the maintenance and equipment control plan, which gives the requirements with respect to frequency of inspection, method of inspection and the acceptance criteria;
- more accurate control of equipment: all instruments are under a documented metrological confirmation plan according with well defined standards (e.g. ISO standards);
- enhanced surveillance program to verify and ensure that the provisions made in the design to ensure safety margins keep on existing and the safety of the plant does not depend upon untested or unmonitored components, systems or structures;
- performance review program aimed to identify and rectify gradual degradation, chronic deficiencies, potential problem areas or causes. This includes review of safety-related non-conformances & failures of SSCs of the plant, determination of their root causes, trend, pattern and evaluation of their safety significance, lessons learnt and corrective measures taken;
- control of documents and records ensures that personnel use only up-to-date documentation avoiding the accidental use of obsolete documents, work instructions or diagrams; also adopting appropriate procedures for the management of obsolete documents. Well managed records ensure a proper process traceability as well as providing a database on the status of the individual components and systems;
- the audit programme, planned and carried out quarterly, takes into consideration the status and importance of the processes and areas to be audited, as well as the results of previous audits. The audit criteria, scope, frequency and methods are documented and are inputs for the periodic management review;
- periodic management review on the status of SSCs: review includes evaluations for improvement opportunities and for changes needed, including quality policy and quality objectives. Records from management reviews are maintained and utilized as input for further management review meetings. The organization applies a specific documented procedure for the conduct of management reviews;
- supply chain: the control of supplies provides an accurate control of purchased products and their suppliers, ensuring compliance with the requirements in terms of safety, quality and reliability. A key point is ensuring adequate availability of spare parts and the implementation of a purchase plan that can compensate for the obsolescence of components, ensuring that changes in technology are compatible with the installed components.

Implementation of an ageing management programme

Based on the positive outcomes from the implementation of the above mentioned activities and for a better and more formal consideration of ageing concerns, LENA has implemented a formal ageing management programme (AMP) following the IAEA guideline SSG-10 “Ageing Management for Research Reactors” in 2014.

The ageing management program result in a set of organized policies, processes, procedures and activities for managing the aging of SSCs with the main purpose of ensuring reliability and availability of required safety functions throughout the extended life of the plant. In order to implement the ageing management programme, the first task was to compare the IAEA guideline with the IMS in place. The outcome leaded to the drafting of a first road map for all the activities not already included or providing adjustments aiming to incorporate ageing in the main management system.

02.3.2 Ageing assessment

WENRA I2.1 The licensee shall assess structures, systems and components important to safety taking into account relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.

In the TRIGA RC-1 reactor ageing mechanisms affecting SSCs relevant to safety are addressed in the implementation of the surveillance rules connected to the conditions and technical specifications attached to the licence. Some specific checks have been also performed in connection to the preparation of the five years report on the status of conservation of the installation. The licensee is planning a prolonged shutdown period in the coming months to perform important maintenance activities and to implement some upgrading measures relevant to safety. A formalized AMP will be established after the completion of the above mentioned activities.

For TRIGA MARK II a central role in the definition of the AMP is played by the categorization of relevant structures, systems and components susceptible to ageing played (figure 1):

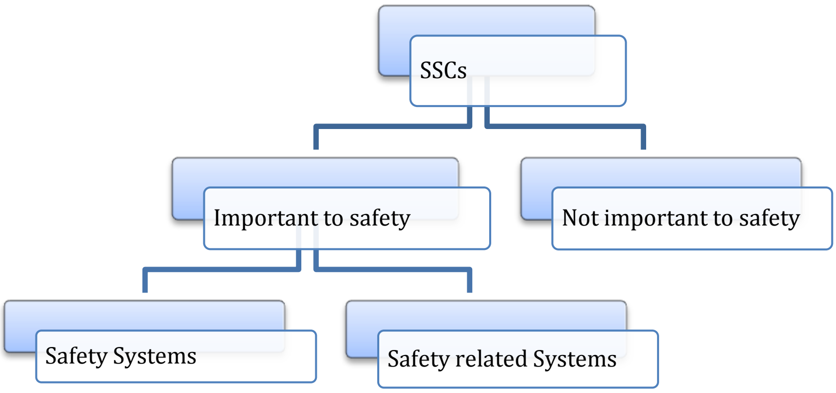

Figure 1: Categorization of structures, systems, components

SSCs were analysed in detail through a series of screenings aiming to identified the most relevant activity areas not already included in the IMS and in need for the implementation of new or revised procedures for ageing management.

Due to the technical nature of the systems and based on specific analysis of past activities results, the following mechanisms have been identified as the most relevant to ageing and therefore to be considered for the implementation of the AMP.

Table 5 – TRIGA MARK II Ageing Mechanism

Ageing mechanism	Code (IAEA SSG 10)
Changes of properties due to neutron irradiation	1
Motion, fatigue or wear (resulting from cycling temp., flow, vibrations, etc.)	4
Corrosion	5
Chemical processes	6
Changes of technology	8
Obsolescence of documentation	10

The following table shows the planned activities regarding the SSCs LENA identified as important to safety, not already included in the periodic maintenance or test & inspection activities, was considered under the AMP and the IMS.

Table 6 – TRIGA MARK II - Planned activities on SSC

SSC	**Planned activities**
Pool structure and vessel	Visual inspections with underwater camera, development of procedures and definition of acceptance criteria. Assessment of results
Core Structure	
Reflector	
Shielding	
Beam Tubes	

SSC	Planned activities
Liner	
Fuel assemblies and storage in reactor pool	
Primary	Efficiency monitoring by on line data acquisitions and real-time parameters evaluation. Trending of data to assess conditions. Periodic result evaluation.
Biological shield	Efficiency of shielding of gamma and neutron dose to be tested every 5 years
Ventilation: emergency	Improved maintenance and controls on the rotating equipment. Definition of procedures
Control Console (LOG channel, SCRAM loops)	New channel refurbishment. Updating of the documentation, management of spare parts. New calibration procedure
Cabling (control console internals and interconnections)	Step by step cable replacement.
Shielding	Visual inspections. Definition of test to be carried out on periodic base.
Beam tube lines	

The ageing management programme therefore include the main following elements:

- screening of Systems, Structures and components identification for reliability and safety Detection, monitoring and trending of ageing degradation
- preventive actions to minimize expected ageing degradation
- continuous improvement of the ageing management program
- management of obsolescence.

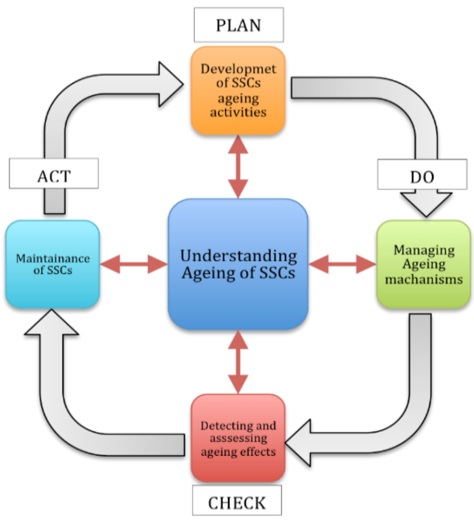

Figure 2

PDCA cycle for ageing management

The overall graded and systematic approach to the reactor ageing management is based on the well known Deming Cycle, as shown in the Figure 2, aiming to the continuous improvement of safety efficiency and effectiveness of the AMP.

02.3.3 Monitoring, testing, sampling and inspection activities

WENRA I2.2 The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service.

As far as concerns TRIGA RC-1, during the activities related to the TRADE experience conducted at the beginning of 2000 , the status of reactor vessel has been tested to evaluate its integrity and to identify possible aging phenomena. The checks were carried out both on the aluminium container and on the liner, as well as on the bellows of the radial channel A and tangential channel.

As concerns TRIGA MARK II the following tables show the monitoring, testing, sampling and inspection activities implemented in the plant.

SSC	Replacement complexity	Ageing mechanism	Periodic checks, preventative or mitigating actions	Frequency
Liner and inner Reactor structures				
Tank structure, liner	A	1,5	Video inspection, ordinary cleaning	Each 3 years
Core structure	A	1,5	Video inspection, ordinary cleaning	Each 3 years
Riflector	A	1,5	Video inspection, ordinary cleaning	Each 3 years
Control rods	A	1,5	Video inspection, ordinary cleaning	Each 3 years
Handling of control rods				
Cabling	B	2,4	Visual Inspection, Regular Cleaning, Greasing, Screw Clamping and Connections	Six moth
Electrical components	B	2,4		Six moth
Mechanical components, leak inspections, wear, tightening connections and joints	B	2		Six moth
Rods guide	B	2		Monthly
Experimental Channels (A, B, C, D, CT, thermalisation column)	A	1,5,7	Video inspection, ordinary cleaning	Each 3 years
Fuel	B	1,3,5	Visual inspection, and/or with camcorder check for elongation and bending	yearly/a third fuel
Fuel storage systems in the tank	C	5	Video inspection, ordinary cleaning	Each 3 years
Measuring Channels (inside the tank)	B	1,5,7	Video inspection, ordinary cleaning	Each 3 years
Core irradiation channels (CC LS rabbit, thermal)	B	1,5,7	Video inspection, ordinary cleaning	Each 3 years
Water quality (tank and pool)	--	--	PH/conductivity measures	Monthly
Physical confinement				
Structure of the Reactor hall	A	--	Internal visual inspection	Daily
			General cleaning	Biannual
			General visual inspection, including external base and cover (sheathing, waterproofing, plastering and bearing structures)	Annual
Concrete shield	A	--	Internal visual inspection	Daily
			General cleaning, doors inspection, channels	Annual
			Neutron map	5 years
Ventilation	B	--	System operation, depression check	daily
			Check flow	Monthly

SSC	Replacement complexity	Ageing mechanism	Periodic checks, preventative or mitigating actions	Frequency
			Engine Verification and Transmission	OLM application
			Instrumental HEPA filtration efficiency check	Annual
Emergency ventilation	B	--	System operation, start test	Monthly
			Check flow	Monthly
			Engine Verification and Transmission	OLM application
			Instrumental verification of active carbon filtration efficiency	18 month
Instrumentation and control				
Control Console	A	4,8,9	Operability verification	Each utilization
			Instrumentation calibration	Biannual
Monitoring area	B	4,8,9	Operability	Daily
			Calibration on site	Three-month
Radiation protection fixed instrumentation	B	8,9	Operability	Daily
			Calibration on site	Three-month
Environmental monitoring units, particulate	C	4,8,9	Operability	Daily
Particulate monitoring reactor hall	B		Calibration on site	
			Operability	Daily
Cooling system reactor	B	4,5,7,8,9	General visual inspection and sealing	Daily
			Pump operability	Monthly
			Conductivity verification	Each utilization
			Measurements and operating data acquisition	Each utilization
			Check filters	
Hydraulic auxiliary reactor systems (pool pump, demineralised water production, water tank)	B	4,5,7,8,9	General visual inspection and sealing	Daily
			Pump operability	Each utilization
Alarm and evacuation systems	B	8	Operability	Daily
Cables	B	2,8	visual inspection	Semestrale
Remote monitoring	B	8		
Pneumatic systems	B	4,8	Operability, lubrication	Da definire

SSC	Replacement complexity	Ageing mechanism	Periodic checks, preventative or mitigating actions	Frequency
Seismic Detector	not installed	-	-	-
Auxiliary equipment				
Electrical and mechanical systems	C	2,8,9	Ordinary maintenance for CPI, work safety regulations	Monthly/biannual/ three years
Generator and UPS	C	2,8,9	Operability, battery test, levels	fortnightly
Fire detection system	B	2,8,9	Scheduled maintenance for Law/NS	Biannual
Fire estinguishing	C	2,8,9	Statutory maintenance (Required by law)	Biannual
Emergency communications (radio, tel, etc)	C	2,8,9	Operability, Battery Testing, Communication Testing	fortnightly
Crane	A	8,9	Visual Inspection, Operability, Functional Testing	three years ISPESL
Radioactive waste storage	B	5	Visual Inspection, Radiation Protection Inspection	Biannual
Well fuel	A	5	Video inspection with camera, ordinary cleaning	Annual
Compressors	C	3,4,8,9	Transmission inspection, pressure gauge, lubricating levels	Annual
Laboratory Hoods	B	4,8,9	Check operability, depression, held	Each utilization
			Instrumental tests	Annual
Documentation				
Documentation facility	B	9,10	Document verification	Annual
Documentation SGQ	C	9,10	Internal Audit	Annual
Operating license	A	9,10		Five years

Table 7 – TRIGA MARK II Monitoring, testing, sampling and inspection activities

02.3.4 Preventive and remedial actions

WENRA I3.2 Surveillance of major structures and components shall be carried out to timely detect the inception of ageing effects and to allow for preventive and remedial actions.

In the past years, many activities were carried out in order to prevent, manage and mitigate ageing effects.

TRIGA RC-1 Preventive actions

Aside from mentioning ordinary maintenance work over the years, mainly related to the replacement of faulty components, the most significant operations to be mentioned are the replacement of components due to ageing identified as result of a continuous process of ageing evaluation.

In the years between 1982 and 1991, these interventions essentially involved equipment of the control system in order to keep electronics at the highest level of efficiency and functionality. During the same period, the following new components were installed:

- twelve environmental monitoring stations in place of outdated ones;
- a meteorological unit with sensors on the top of the building;
- a concentrator of data from all monitoring stations;
- a data acquisition system for storing all checked parameters.

Upgrading interventions

There have been no significant upgrading interventions during the period 1991-2005, except the following:

- **2000-2001:** Structural verification of the plant for subcritical configuration for TRADE Experience.

Years 2005–2010:

- Logarithmic Amplifier Replacement;
- Process Temperature Indicators Replacement;
- replacement of old recorders with 3 ABB Videographic Recorders**.**

Years 2010–2015:

- replacement of 2 fuel temperature indicators;
- replacement of the old ion exchange demineralization plant with a new reverse osmosis demineralized water production plant;
- replacement of the environmental monitoring system (outdated components become no longer available);
- replacement of the secondary console racks in the Control Room because obsolete and no longer able to guarantee the correct ventilation;
- replacement of the water circulation pump for Experiences Pool due to the seals failure.

It is to be mentioned that in 1999 the transfer of irradiated fuel elements to the United States of America, as country of origin, took place.

TRIGA Mark II - Preventive actions

The most relevant under safety, technical and economical point of view were focused on:

Biological Shield efficiency test (1997):

After about 35 years of utilization of the biological shield a new neutron and gamma dose inspection has been performed in order to verify the shielding efficiency. The 1997 campaign of measurements was made with modern and calibrated instruments. Gamma dose and neutron dose have been assessed separately on each wall of the biological shield. The tests showed that there were no changes compared to the original situation. Based on the results of the tests, changes and implementation of safety procedures have been introduced, in particular in the field of radiation protection for some mechanical and electrical maintenance procedures.

Electrical power distribution and emergency power supply (2000)

The electrical system was completely replaced due to the obsolescence of components, changes in regulations and cable deterioration. Even if not strictly classified as a safety system, its availability and reliability is important to assure a correct operation of the reactor. Furthermore, the installation of more performing components, together with more efficient surge suppressors, avoids and contains potential damages due to external events. Safety-critical systems are operated as privileged loads, through an on-line type UPS. The emergency power supply was also upgraded by installing a new and a more performing a diesel generator. The new installed components lowered the overall fire-related risk of the electrical power distribution, allowing an overall safety improvement.

Heating, ventilation and air filtration system (2001):

Due to the aging of components, repetitive unscheduled maintenance activities and in order to avoid unplanned long reactor shut-down periods, the new system was completely renewed with a new computerized air filtering and ventilation system. The inlet air is filtered by high efficiency (~ 95%) filters while the air extracted from the reactor room is filtered by absolute HEPA filters (total efficiency > 99.95%) before the release in atmosphere. The air extraction is powered by an inverter-driven motor that keeps automatically the reactor building at 50 Pa less than atmospheric pressure, as required by technical specifications. The new system is now equipped with a modern computerized supervisory system capable of visualizing and trending data, feature resulted particularly useful for preventive maintenance purpose.

Radiation monitoring system (2006):

The area radiation monitoring system has been renewed after 30 years of operation. The new system, based on proportional counters, a micro-computer (PLC) and an home-made software, completely replaced the former one based on analogue components and Geiger type counters. The system collects the data sampled by six β-γ dose-rate proportional counters, a free-air ionization chamber and a weather station through a serial data bus. Collected data are stored and displayed on PC in the reactor control room as well as in the emergency control room. The software allows the operator to access the data, modify parameters and perform tests remotely.

Cooling system (2007):

a complete substitution of the tertiary heat exchanger and a partial substitution of the components of the secondary heat exchanger circuit was realized, due to corrosion and degradation phenomena. In particular, the substitution involved the following components: secondary/tertiary circuit heat exchanger (plate type); tertiary circuit water flow control valve, valve drive motor and valve controller (PID type); tertiary circuit water filters and magnetic filter for water macromolecules and carbonates removal. For predictive and improved maintenance purpose, a dedicated PLC-based data acquisition system was implemented in order to collect data thus obtaining on line parameter for evaluating the system behaviour (e.g. heat exchange efficiency, pumps efficiency, pressure/temperature drops across the exchangers).

Reactor console (2008):

Most of I&C system were in operation since their first installation in 1965. During the years several repairs and had to be carried out, affecting the system availability. Moreover, the original design was based on electromechanical components and vacuum tubes, then the availability of spare parts was becoming a critical issue. In order to grant a safe and continuous reactor operation for the future, improving reliability and extending the system life time, most of the I&C were refurbished with home-made designed system using high quality commercial components. As a result, almost a complete substitution, channel-by-channel, of the I&C system was realized without changing the operating and safety logics. Thanks to the new components, a dedicated acquisition system was implemented for data collection and analysis. System Substituted are: Power Level Channels, Water Radioactivity meter, Water temperature meter, Water conductivity gauges, Peak neutron flow indicator circuit And integrated, Fuel Temperature Indicator Circuit, Automatic power adjustment circuit, HV supply.

The off-gas radiation monitoring system (2009):

it was completely replaced with a new measurement system based on a NaI spectroscopy detector. The gamma-ray spectrum of reactor gaseous effluents is collected on-line using commercial software and data are remotely accessible (i.e. from reactor control room or from reactor emergency control room). In addition, the environmental airborne particulate monitoring system was completely upgraded and redesigned for a better efficiency and reliability.

Water purification system (2010):

After about 20 years of utilization, in the filling-water demineralization system was completely replaced with a new mixed-bed, laboratory-grade demineralizer. In this type of demineralizer, cation and anion resin beds are mixed together, resulting therefore equivalent to a two-step demineralizers in series. In a mixed-bed demineralizer, more impurities are replaced by hydrogen and hydroxyl ions, and the water that is produced is extremely pure. The new system allows a safer and quicker resins replacement avoiding the personnel to deal with resins regenerations and acid/basis solution handling.

Re-evaluation of accident scenarios and review of emergency plans (2010):

A new revaluation of the accident scenarios and its assumptions in the light of new guideline, standards and legislation, was performed leading to the review of the emergency plans. Each year is organized at list an emergency drill in presence of the regulatory body with the cooperation of the responsible institutions. Note and recommendation are discuss in a final briefing.

Data acquisition system (2012):

several efforts have been made recently in order to implement a comprehensive data acquisition system for the most important parameters (e.g. reactor console, radiation monitoring systems, ventilation, cooling systems). The system allows to constantly monitoring parameters, early detecting anomalies or drifting instruments in order to prevent potential and unexpected failures. This can be seen as an extensive predictive maintenance activity, resulting useful also for ageing purpose; in fact, the trend of performance can be a good indicator of system degradation, and the related data analysis can positively contribute to a more efficient understanding of ageing mechanisms and their counter measures.

Video inspection (2014):

A campaign of visual inspection (with miniature TVCC/endoscopes) of the reactor pool internals and spent fuel dry storage has been carried out and is scheduled for future activities. Results have been reported and evaluated within the safety committee during the meeting on 31 October 2014.

Operating personnel training programmes review (2014):

The proposed programmes provide general guidance to meet the training requirements and qualification of the research reactor based on best practices.

Reassessment of external events (2016):

Updated information about siting, demographics, seismicity, hydrology and meteorology

Bridge Crane (2016):

Verification of the crane with operational test.

New particulate monitor

Ambient monitor for beta measurements in the air with two cylindrical stainless steel ionization chambers, integrated electronic processing complete with display and functional interface for the operator

New UNIT 300 - control unit (2016):

the analysis of non-conformity recoded with increased number of SCRAMs related to UNIT 300 (control unit) and the decreasing reliability of Analog components, the lack of spare parts and cables ageing lead to the substitution of this UNIT with a new one with same logic and update components.

New Log Channel (2017):

The intervention is contextualized as a preventive action due to obsolescence of the components and the extreme difficulty of finding spare parts of the system in use. At the present state of the project, the Mirion DWK 250 (optionally DWK 260) channel has been selected, which will replace the previous one, including the detection system (new fission chamber with the same characteristics as that installed and connecting cables) and the preamplifier. This system, which has already been successfully used as a safety circuit in several nuclear power plants, has also had positive feedback from several research reactors including, among others, the ACRR reactor of SANDA Lab - New Mexico, USA - MIT - Boston, USA. The project foresee a test stage with te new system working in parallel with the old one (non connects with the safety circuitry) in order to obtain a robust database for determining the correct functioning of the system and the response of the measurements between the two channels. Only at the end of the test phase, based on positive outcomes from the data benchmark, the channel will be replaced.

02.4 Review and update of the overall AMP

WENRA I2.3 The Periodic Safety Reviews shall be used to confirm whether ageing and wearout mechanisms have been correctly taken into account and to detect unexpected issues.

WENRA I2.4 In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy.

WENRA I2.5 The AMP shall be reviewed and updated as a minimum with the PSR, in order to in-corporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant.

In 2013 ISPRA advised the Ministry of Economic Development to ask the licensees of research reactors to update their safety assessment updating process, taking into account the Code of Conduct recommendations. As result, in the context of the five years report on the status of the reactor a PSR is conducted with included several aspects of an ageing management programme.

In this context it is taken into account that through the Plant Safety Committee, a continuous review of the safety of the plant and of the status of preservation of its SSCs is carried out. Any change of the technical equipment of the installation relevant from the point of view of nuclear safety and radiation protection are documented, communicated and has to follow a specific approval procedure. The Committee is also involved in the safety analysis of new experimental activities and on aspects related to the use and aging of the main systems. Each activity is regularly documented and reported to the Regulatory body. An accurate maintenance plan and monitoring, followed by the operating instructions, is established in accordance with the Technical Specifications and the related surveillance programme.

For TRIGA MARK II reactor the integrated Management System allow the systematic review of the plant operation and is used as input for the Periodic Safety Review. These information are used to confirm whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues.

For TRIGA RC-1 reactor, as already stated, in the next months important upgrading activities of reactor SSCs are planned with the aim of improving the reactor safety. After these interventions, using the information obtained during previous maintenance activities, with the assessment of the state of conservation and operation, together with the results of the Periodic Safety Review carried out over the years, a complete ageing management program will be drawn up following the guidelines reported in the “Topical Peer Review 2017 - Ageing Management Technical Specification for the National Assessment Reports”.

02.5 Licensee’s experience of application of the overall AMP

As far as concern TRIGA RC-1, as already stated, several aspects of an ageing management programme have been implemented in the past years operation leading to the preservation of SSCs operability throughout the implementation of specific maintenance and replacement interventions.

As already described the Licensee has now planned a prolonged shutdown period to implement several important maintenance and upgrading interventions, for which a licensing process is starting.

This shutdown period will be an opportunity for a revision of the status of the relevant SSCs.

When the above mentioned upgrading programme will be completed the Licensee plans to develop a specific AMP that will take into account past experience of operation.

As far as concern TRIGA Mark II, the lessons learned from the AMP implementation are evaluated in the light to introduce further improvements.

02.6 Regulatory oversight process

The regulatory oversight process of research reactors into operation is mainly based on the regular inspection and control activities, licensing process of plant modifications and assessment of five years report.

For Italian research reactors a periodic review is foreseen every 5 years addressed mainly on monitoring the status of installations with analyses based on case by case approach. ISPRA has requested the licence holder to submit an updated evaluation of safety based upon recommendations of the IAEA Code of Conduct.

Implementing the “**Code of Conduct on the Safety of Research Reactors”** to Italian RRs, a more systematic application of the PSR is in progress.

The reviews are include an assessment of the plant design and operation using the Specific Safety Guide, SSG-25 “Periodic Safety Review for Nuclear Power Plants”, as guidance on how to conduct PSRs applied according to a graded approach. WENRA reference levels are also taken into account.

An updated assessment has been presented by all the operating reactors.

The review process of the five years reports is conducted :

- with the short-term goal to improve the management of the safety documentation and to carry out an assessment of the current status of each individual system;

− long-term goal is to cover all safety factors identified in the PSR framework.

02.7 Regulator's assessment of the overall ageing management programme and conclusions

Taking into account the age of the reactors particular attention is devoted from the regulatory point of view to the proper management by the licensees of aging mechanism, either related to physical ageing, leading to the degradation of the physical characteristics of SSCs, or to non-physical ageing related to technological obsolescence.

As far as concern TRIGA Mark II, it is recognized that an Integrated Management System is implemented in accordance with International Standard ISO 9001. On that basis an ageing management programme (AMP) is also in place. The outcome of the comparison of the IAEA guideline SSG-10 "Ageing Management for Research Reactors" with the IMS in place leaded to the drafting of a first a road map for all the activities not already included or providing adjustments aiming to incorporate ageing in the main management system whose application is monitored by ISPRA.

With regard to TRIGA RC-1 , ISPRA has taken note an Ageing Management Program of the reactor has been prepared by the licensee and it is expected to be implemented at the end of the program of interventions on infrastructures of the reactor building, on the electrical power supply and on the instrumental control. The licensing application for these interventions is expected to be submitted in the coming months.

ISPRA has taken the decision to request the Licensee the conduct of the first inspection activities identified in the Ageing Management Programme for Electrical Cables, Hydraulic Circuit and Container in connection with the envisaged prolonged shutdown phase in 2018.

03

Electrical cables

-

03.1 Description of ageing management programmes for electrical cables

03.1.1 Scope of ageing management for electrical cables

TRIGA RC-1

Within the TRIGA RC-1 system, as shown in appendix 2, the operator identified only two categories of electrical cables to which refers the WENRA document: low voltage cables (V <380 Volt) and signal cables (neutron flow measurement).

For these types of cables the licensee has expressed the intention to introduce the evaluation of the ageing with respect to the following characteristics:

- Conduction;
- Insulation;
- Armature/shielding;
- Coating;
- Endings/contacts.

For the electrical cables, implementation of methodologies and procedures for the assessment of the ageing is being carried out following a gradual approach according to the IAEA SSG-10 guidance. In parallel and in view of a complete renewal of the control equipment and power supply system, a step-by-step replacement process of the electronic and electrical components is being carried out, leaving unchanged the layout and control logic of the reactor.

TRIGA MARK II

The electrical cables selected within the scope of the ageing management are that relevant to the reactor safety. In consideration of the probability and consequence of a system malfunctions, related to electrical cables, and their effects are those directly connected to the rod control and the power measure. This evaluation take in account the possible scenarios for each malfunctions or combination of them; whereas the control and monitoring of nuclear and power parameters can affect the reactor operation itself, the other system are more related to the reactor reliability.

Electrical cables can be group into three categories:

- High voltage cables
- Signal cable
- Internal cables

These groups finally are part of two major system:

- Control Console (LOG channel, SCRAM loops)
- Cabling (control console internals and interconnections)

03.1.2 Ageing assessment of electrical cables

The ageing mechanisms identified by the licensees are:

1 -Properties changing (physical, chemical, mechanical) due to neutron irradiation;

6- Chemical processes;

2-Fatigue phenomena, material consumption due to thermal cycles, mechanical load, flow, induced vibrations

8-Change of technology

9- Regulatory rules changes (regulations, technical specifications, etc.).

(codes according to IAEA SSG10)

Acceptance criteria are defined taking into account the advice of the Plant Safety Committee, in consideration of OLC's, review by the internal quality assurance with performance indicator analysis, International standards (IAEA, EURATOM) and external peer reviews .

The replacement for control rods, cables, connections and insulators is normally considered of *high* complexity and for SSC and measure/safety detectors *of average* complexity.

03.1.3 Monitoring, testing, sampling and inspection activities for electrical cables

TRIGA RC-1

The low voltage and signal electrical cables inside the system are mainly those installed when the reactor was built. Although no formal implementation of the ageing management program has been executed over the years, due to the periodic inspections of the plant operating procedures the proper functioning of the electrical components, even indirectly, has been verified.

Taking into account the ageing mechanism and the components important for safety the licensee has proposed a possible ageing evaluation program, summarized in the following table, in which for each component inspection type, mechanism of ageing and replacement criticality are described.

Table 8 –TRIGA RC-1 - Ageing evaluation program for electrical cables and electronic

Component	Inspection type	Mechanism of ageing[2]	Replacement criticality
Power and control cables of cooling system	Functional instrumental	2, 6, 9	B
Power and control cables of ventilation system	Functional instrumental	2, 6, 9	B
Power and control cables of instrumentation (console, detectors, experimental facilities	Functional instrumental	2, 6, 9	B
Power and control cables of services (switchboards, crane, lights etc.)	Functional instrumental	2, 6, 9	B
Power and control cables of I&C systems (console, Control rods drive mechanisms	Functional instrumental	2, 6, 9	B
Instrumentation and control cables of radiation protection system	Functional instrumental	2, 6, 9	B
Signal cables of neutron flux measurement system (power measurement channels)	Functional instrumental	1, 2, 6, 9	A
Signal cables of radiation protection controls system.	Functional instrumental	2, 6, 9	B

A –High; B -Medium

TRIGA MARK II

Every utilization of the SSC's a check list of test regarding the systems operability is filled. Any information relevant for the reliability is advised with a non-conformance. Each week or month, the operability is tested also with the help of a neutron source and the nuclear parameter are taken and registered. Acceptance criteria are set in the OLC's and trend in values variation and progressive deterioration are advice to the supervisors and Technical Director. This information is discussed in the Plant Safety Committee.

It is scheduled, each six month, an overall SSC's check. Results are recorded in a register and presented to the Plant Safety Committee together with the report on the SSC's ageing situation.

Next table summarized the activities undergoing in the AMP.

[2]The meaning of the values for the Ageing Mechanisms and Replacement Criteria fields is described in the Annex 2

Table 9 –Triga MARK II - Ageing evaluation program for Control console and cabling

SSC	Replacement complexity	Ageing mechanism	Periodic checks, preventative or mitigating actions	frequency
Control Console	A	4, 8, 9	Operability tests	Each utilization
Cabling	B	2, 4	Visual Inspection, Regular Cleaning, Greasing, Screw Clamping and Connections	Six moth

A –High; B -Medium

03.1.4 Preventive and remedial actions for electrical cables

TRIGA RC-1

The renovation and replacement of some panels and the consequent installation of new cables, e.g.: in the case of the installation of the new water demineralizer, of the new environmental monitoring equipment, of the new auxiliary console cabinets, etc., was a consequence of a modernization process for component obsolescence and not of a formal replacement program related to ageing.

The implementation of an ageing management program will allow future replacements and upgrades, mainly taking into account the parameters strictly related to ageing.

TRIGA MARK II

Many activities have been carried out in order to prevent, manage and mitigate aging effects on Electrical cables. In 2008 In order to grant a safe and continuous reactor operation for the future, improving reliability and extending the system life time, most of the I&C were refurbished with home-made designed system using high quality commercial components. As a result, almost a complete substitution, channel-by-channel, of the I&C system was realized without changing the operating and safety logics. System substituted are: Power Level Channels, Water Radioactivity meter, Water temperature meter, Water conductivity gauges, Peak neutron flow indicator circuit and integrated, Fuel Temperature Indicator Circuit, Automatic power adjustment circuit, HV supply. During these activities, most of the cables have been replaced.

In 2016, the analysis of non-conformity recoded with increased number of SCRAMs related to UNIT 300 (control unit) and the decreasing reliability of analogical components, the lack of spare parts and cables ageing lead to the substitution of this UNIT with a new one with same logic and update components.

Finally, as preventive action a new logarithmic channel has been selected, which will replace the previous one, including the detection system (new fission chamber with the same characteristics as that installed and new connecting cables) and the preamplifier. This system, which has already been successfully used as a safety circuit in several nuclear power plants, has also had positive feedback from several research reactors including, among others, the ACRR reactor of SANDA Lab - New Mexico, USA - MIT - Boston, USA. The project foresee a test stage with the new system working in parallel with the old one (non connects with the safety circuitry) in order to obtain a robust database for determining the correct functioning of the system and the response of the measurements between the two channels. Only at the end of the test phase, based on positive outcomes from the data benchmark, the channel will be replaced. This action is due to the obsolescence of the components and the extreme difficulty of finding spare parts of the system in use.

Heretofore, around 80% of the cables mentioned in par 3.1.1 have been replaced, including those related to the radiation area monitoring system (2006). Upon the completion of the test and commissioning stage of the aforementioned logarithmic channel, the remaining I&C wirings (mostly interlinks between the control console drawers) will be replaced.

The following table summarizes the next steps in the implementation of the AMP.

SSC	Next steps
Control Console (LOG channel, SCRAM loops)	New channel refurbishment. Updating of the documentation, management of spare parts. New calibration procedure
Cabling (control console internals and interconnections)	Step by step cable replacement.

As far as other cables relevant to safety and non-safety such as the normal power supply to the, all cables have been replaced during the replacement of the power supply system in the early 2000 facility. Moreover, the main line and the medium to low voltage cabin feeding the reactor building has been replaced in 2008. All cables related to the power supply undergo to the general periodic maintenance plan according with Italian laws, standards and regulation (electrical and fire safety). These include periodic verification of cable insulation, continuity test, grounding tests, visual inspection.

03.2 Licensee’s experience of the application of AMPs for electrical cables

TRIGA MARK II

The Licensees acknowledge that the integrity of cables and connectors is vital for the safe and efficient long-term operation of the facility and has put in place the AMP to proactively deal with safety and performance concerns. The application of the AMP related to cables has to take into consideration several factors in order to grade and optimise resources and efforts on this subject. Mayor factors concurring in the definition of the extension of the cable ageing management are:

- Total number of cables
- Number of cables under permanent irradiation
- Safety functions and safety-related function related to the cables
- General maintenance practices already in place
- Average age of cables
- Ease of replacement

- Regulations and legal framework in place
- Allocated annual budget.

Based on the above, an approach focused on an improved monitoring and assessment system rather than the application of quantitative techniques has been adopted. In fact, while the market offers - though mostly for the NPPs business - specific tools and highly specialised techniques (e.g. reflectometry) to assesses the remaining life or "qualified life", for a small sized facility such as LENA's, where the replacement is not such an obstacle factor, a more frequent replacement of safety cables can be easily afforded. Furthermore, it has be taken into account that the working conditions of cables in a typically small type TRIGA reactor are not quite demanding in terms of temperature, mechanical and radiation stress.

More in detail, the following considerations should be highlighted:

- Safety system cables in core can be difficult to replace but the cable+detector electrical resistance is measured and recorded every 6 months. Trends are evaluated in order to detect possible degradation phenomena;
- Out of core, cables and connectors are periodically inspected (visual inspection), and cable+detector electrical resistance is measured and recorded every 6 months. Trends are evaluated in order to detect possible degradation phenomena. In this event, a case by case solution is discussed and evaluated in order to find the most suitable corrective action to be implemented;
- In consideration of the low electrical, thermal and mechanical stress on cables, the major concern is related to the insulating and jacket materials that are subjected to a time-dependant degradation. Since cable trays are easily accessible, jacket conditions can be simply checked for creeps or other mechanical damage. In this case, cables are promptly replaced;
- Finally, the utilization of nowadays polymers and mineral insulated cables will significantly lengthen the cables average life, which can be conservatively assumed of at least 20 years.

03.3 Regulator's assessment and conclusions on ageing management of electrical cables

For TRIGA RC-1 ageing problems have been addressed throughout the systematic implementation of the surveillance rules connected to the conditions and technical specifications attached to the licence of the reactor. Important extraordinary maintenance and upgrading interventions are planned in 2018. A specific licensing process will be conducted for above mentioned interventions.

A formalised ageing management program has been prepared to be applied following the implementation of above mentioned interventions.

ISPRA has taken the decision to request the Licensee the conduct of the first inspection activities identified in the Ageing Management Programme for Electrical Cables, in connection with the envisaged prolonged shutdown phase in 2018.

For TRIGA MARK II an ageing management program has been implemented based to the experience gained with the Integrated Management System and is considered adequate.

04

Concealed pipework

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04.1 Description of ageing management programmes for concealed pipework

04.1.1 Scope of ageing management for concealed pipework

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TRIGA RC1

The TRIGA RC-1 is equipped with three separate hydraulic circuits (Primary, Secondary, Derived) each made of different materials and wall thicknesses. Aluminum for Primary, Carbon Steel for Secondary and PVC for Derived. All pipes are visible, inspected externally every day. The wall thickness of the piping results to be clearly oversized with respect to the operating pressures as well as the process temperature value being very low. During the years of operation, no leakage phenomena from junctions or even from pipes have been shown.

TRIGA MARK II

In the management program the reactor cooling system and associated systems are included. There are not concealed pipework and visual inspection can be carried out daily. The management program include a periodical verification of hydraulic efficiency.

04.1.2 Ageing assessment for concealed pipework

The ageing mechanisms identified by the licensees are:

4 - Fatigue phenomena, material consumption due to thermal cycles, mechanical load, flow, induced vibrations;

5- Corrosion;

6- Chemical processes;

7- Erosion;

8- Technology change;

9- Regulatory changes, regulations, technical specification s, etc.

(codes according to IAEA SSG10)

Acceptance criteria are defined in consideration of the comparison between historical data and new data.

The operating experience gained during the years allows to state that the reactor hydraulic circuit, due to the not particular harsh operating conditions (low pressures, low temperatures, moderate flows), does not show any obvious phenomena of degradation attributable to ageing. The water quality is one of the parameters that can have major consequences. Its quality is however well checked in order to minimize ageing mechanism.

It has also to be considered that the complexity for components replacement interventions is considered low.

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04.1.3 Monitoring, testing, sampling and inspection activities for concealed pipework

TRIGA RC1

Taking into account the ageing mechanism and the components important for safety the licensee has proposed an ageing evaluation program, summarized in the following table, in which for each component inspection type, mechanism of ageing and replacement criticality are described:

Table 10 –Triga RC-1 - Ageing evaluation program for hydraulic circuit

Component	Inspection type	Mechanism of ageing	Replacement criticality
Primary Circuit	Visual	4, 5, 6	A
	Instrumental		
Secondary Circuit	Visual	4, 5, 6	A
	Instrumental		
Purifying Circuit	Visual	4, 5, 6	C
	Instrumental		

A – High; C – Low

TRIGA MARK II

The TRIGA is equipped with three separate hydraulic circuits (Primary, Secondary, Derived) which are monitored with a daily overall visual inspection. Other operating parameters are acquired and in check list. An overall review is discussed in the Plant Safety Committee.

The activities undergoing in the AMP are summarized in the following table.

Table 11 –Triga MARK II - Ageing evaluation program for reactor cooling system

SSC	Replacement complexity	Ageing mechanism	Periodic checks, preventative or mitigating actions	frequency
Instrumentation and control				
Cooling system reactor	B	4, 5, 7, 8, 9	General visual inspection and sealing	Daily
			Pump operability	Monthly
			Conductivity verification	Each utilization
			Measurements and operating data acquisition	Each utilization
			Check filters	
Hydraulic auxiliary reactor systems (pool pump, demineralised water production, water tank)	B	4, 5, 7, 8, 9	General visual inspection and sealing	Daily
			Pump operability	Each utilization

B -Medium

04.1.4 Preventive and remedial actions for concealed pipework

TRIGA MARK II

Many activities were carried out in order to monitor the efficiency of the hydraulic system and assure a reactor good performance in terms of reliability and availability. In 2008, a complete substitution of the tertiary heat exchanger and a partial substitution of the components of the secondary heat exchanger circuit was realized, due to corrosion and degradation phenomena. In particular, the substitution involved the following components: secondary/tertiary circuit heat exchanger (plate type); tertiary circuit water flow control valve, valve drive motor and valve controller (PID type); tertiary circuit water filters and magnetic filter for water macromolecules and carbonates removal.

The following table summarizes the implementation of the AMP.

SSC	Planned activities
Primary	Efficiency monitoring by data acquisitions and real-time parameters evaluation. Trending of data to assess conditions. Periodic result evaluation.

04.2 Licensee’s experience of the application of AMPs for concealed pipework

TRIGA RC1

The experience gained during the years of operation allows to state that the reactor hydraulic circuit, due to the not particular burdensome operating conditions (low pressures, low temperatures, moderate flows and good water quality), does not show any obvious phenomena of degradation attributable to ageing.

Following the planned upgrading programme in 2018 the Ageing Management Programme under development will cover also reactor piping.

TRIGA MARK II

The Licensees acknowledge that water quality is the base of the pipework good preservation. The application of the AMP is related to perform the ordinary maintenance and periodically review of the operating parameters are acquired.

04.3 Regulator’s assessment and conclusions on ageing management of concealed pipework

It is considered that all the pipes of circuits relevant to safety are visible to inspection. In addition, reactors operating conditions are characterized by low pressures, low temperatures, moderate flows which do not represent particularly challenging conditions for relevant piping. The preservation of good water quality conditions is also ensured.

For TRIGA RC-1 ageing problems have been addressed throughout the systematic implementation of the surveillance rules connected to the technical specifications of the reactor. Important extraordinary maintenance and upgrading interventions are planned in 2018. A formalised ageing management program is under preparation and will start to be applied in connection to these interventions. ISPRA has taken the decision to request the licensee to conduct the inspection activity envisaged in the programme for hydraulic circuits in connection with the prolonged shutdown of the reactor for extraordinary maintenance and upgrading measures implementation, planned for 2018.

The AMP in place for TRIGA MARK II is considered adequate.

05

Reactor vessels

05.1 Description of ageing management programmes for RPVs

05.1.1 Scope of ageing management for RPVs

TRIGA RC1

During the activities related to the TRADE experience conducted in 2000, the status of reactor vessel has been tested to evaluate its integrity and to identify possible ageing phenomena. The checks were carried out both on the aluminium container and on the liner, as well as on the bellows of the radial channel A and tangential channel.

As already mentioned in section 2.3.3, these surveys, carried out using ultrasonic techniques, have allowed to verify the status of components subject to particular ageing agents such as neutron and gamma irradiation as well as corrosion phenomena.

TRIGA MARK II

Based on the preliminary aging management program for the TRIGA Mark II Reactor, a periodical video inspection of the reactor tank is foreseen. The tank is identified as the structure component essential for the Reactor safety in consideration of the probability and consequence of its damage. The video inspection scope is to detect initial oxidation states on mullite with the purpose of capturing an image database for subsequent comparison checks and for checking the parts not directly visible from the liner's top.

05.1.2 Ageing assessment of RPVs

The ageing mechanisms identified by the licensees are:

5 - Corrosion;

6-Chemical processes;

(Codes according to IAEA SSG-10)

Acceptance criteria are defined, within the Plant Safety Committee, in consideration of the comparison between historical data.

05.1.3 Monitoring, testing, sampling and inspection activities for RPVs

TRIGA RC1

Taking into account the ageing mechanism and the component important for safety the licensee has proposed an ageing evaluation programme, summarized in the following table, in which inspection type, mechanism of ageing and replacement criticality are described:

Table 12 –Triga RC-1 - Ageing evaluation program for container

Component	**Inspection type**	**Mechanism of ageing**	**Replacement criticality**
Liner	Visual	5, 6	A
	Instrumental		

A – High

TRIGA MARK II

The liner is divided into 13 vertical sections plus the top of the fuel and reflector grid. The chamber, when positioned at the top or bottom edge of the liner, moves at a speed sufficient to capture a sharp image along the surface, taking care to maintain a sufficient distance to avoid blinding due to reflector illumination. Shooting takes place in those parts of the section where there are points that are considered potentially critical due to the channels, the thermal channel or thermal column. Once on the bottom of the tank, the chamber is rotated 180 ° to allow a view of the reactor components such as the core, the reflector and the thermal column. The 13 sectors are marked on top of the reactor and capture files take the corresponding numbering. Any information relevant for the reliability is advised with a non-conformance.

It is scheduled, every 3 years, an overall SSC's check. Results are recorded as MPEG files on USB storage and presented to the Plant Safety Committee, together with the report on the SSC's ageing situation.

The following table summarizes the implementation of the AMP.

Table 13 –Triga MARK II - Ageing evaluation program for reactor tank

SSC	**Replacement complexity**	**Ageing mechanism**	**Periodic checks, preventative or mitigating actions**	**frequency**
Tank structure, liner	A	1,5	Video inspection, ordinary cleaning	Every 3 years

A - High

05.1.4 Preventive and remedial actions for RPVs

TRIGA MARK II

Many activities were carried out in order to prevent, manage and mitigate aging effects on reactor containment. To minimize corrosion the water quality is daily checked and it is avoid the utilization of other materials except for aluminium. For these reasons, in 2010, after about 20 years of utilization, the filling-water demineralization system was completely replaced with a new mixed-bed, laboratory-grade demineraliser. The water produced is extremely pure. In 2014, a campaign of visual inspection (with miniature TVCC/endoscopes) of the reactor pool internals and spent fuel dry storage has been carried out and is scheduled every 3 years. Results are reported and evaluated by the Plant Safety Committee. The tasks included in the control plan are the periodic inspection, the suction of residues deposited at the bottom of the tank with the recovery of fallen objects and a general aspiration with surface filtration.

The following table summarizes the implementation of the AMP

SSC	Planned activities
Pool structure and vessel	Visual inspections with underwater camera, revision of procedures and acceptance criteria. Assessment of results
Core Structure	
Reflector	
Shield	
Beam Tubes	
Liner	
Fuel assemblies and storage in reactor pool	

Maintaining the physical chemical characteristics of the primary circuit water has mitigated the aging of the structure.

05.2 Licensee's experience of the application of AMPs for RPVs

TRIGA MARK II

The Licensees acknowledge that the integrity of reactor containment is vital for the safe operation and has been put in place the AMP to proactively deal with safety and performance concerns. The application of the AMP related to tank corrosion has to take into consideration the water quality and materials inserted in the inner.

It can be said that the tank of the TRIGA MARK II reactor, installed at the LENA Centre of the University of Pavia has been properly maintained over the years and has no particular heterogeneity. The only reliefs are highlighted and are subject to periodic verification by comparing the new images with the database created.

05.3 Regulator's assessment and conclusions on ageing management of RPVs

For TRIGA RC-1 Reactor in 2000 the status of reactor vessel has been tested to evaluate its integrity and to identify possible ageing phenomena. These surveys, carried out using ultrasonic techniques, have allowed to verify the status of components subject to particular ageing agents such as neutron and gamma irradiation as well as corrosion phenomena.

As said, the Licensee is planning an extraordinary and maintenance upgrading programme starting next year. In view of the operation of the reactor after the implementation of this programme an AMP is prepared, including also aspects related to liner integrity. ISPRA has taken the decision to request the licensee to conduct the inspection activity envisaged in the programme for container in connection with the prolonged shutdown of the reactor for extraordinary maintenance and upgrading measures implementation, planned for 2018 .

For TRIGA MARK II it is recognized that many activities were carried out in order to prevent, manage and mitigate ageing effects on reactor containment. To minimize corrosion the water quality is daily checked and the utilization of other materials except for aluminium is avoided. For these reasons, in 2010, after about 20 years of utilization, the filling-water demineralization system was completely replaced with a new mixed-bed, laboratory-grade demineraliser. The water produced is extremely pure. In 2014, a campaign of visual inspection was also conducted. The approach followed by the licensee is considered adequate.

06

Concrete containment structures

(Chapter 7 WENRA)

06.1 Description of ageing management programmes for concrete structures

06.1.1 Scope of ageing management for concrete structures

TRIGA RC1

A re-assessment of the seismic resistance of the reactor building structures is on going in the context of the licensee programme extraordinary maintenance and upgrading of the reactor. In this context also the status of other concrete structures will be assessed.

TRIGA MARK II

In the management program, two main concrete part are considered as containment: the biological shielding and the building of the reactor hall. The management program include a periodical verification of the shield behaviour and the building structure was re-evaluate against the new seismic classification of the area.

06.1.2 Ageing assessment of Reactor containment

TRIGA MARK II

Ageing mechanisms are the change of properties (physical, chemical, mechanical) due to exposition at the environmental forces along the reactor life and the change in normative and references.

Acceptance criteria are defined, within the Plant Safety Committee, in consideration of the comparison between historical data.

The replacement complexity is considered high.

06.1.3 Monitoring, testing, sampling and inspection activities for Reactor containment

TRIGA MARK II

The biological shield is monitored each day with comparison to dose values at determined sampling points and with a daily overall visual inspection of the reactor building. Due to the change of seismic classification zone and the safety reassessment in the light of the Fukushima Daiichi nuclear power plant accident, the reactor building has been re-evaluated against external events and mostly against new seismic data for the area of installation.

Reassessment results together with the check list are presented to the Plant Safety Committee.

The following table summarizes the activities undergoing under the AMP.

Table 14 –Triga MARK II - Ageing evaluation program for concrete containment structures

SSC	***Replacement complexity***	***Ageing mechanism***	***Periodic checks, preventative or mitigating actions***	***frequency***
Physical confinement				
Structure of the Reactor hall	A	--	Internal visual inspection	Daily
			General cleaning	Biannual
			General visual inspection, including external base and cover (sheathing, waterproofing, plastering and bearing structures)	Annual
Concrete shield	A	--	Internal visual inspection	Daily
			General cleaning, doors inspection, channels	Annual
			Neutron map	5 years

A - High

06.1.4 Preventive and remedial actions for concrete structures

TRIGA MARK II

Many activities were carried out in order to monitor and identify ageing effects on concrete containment. To assure a continuous monitoring the biological shield is daily checked and ordinary building maintenance is planned. In 2016 information about siting, demographics, seismicity, hydrology and meteorology was updated. Results are reported and evaluated within the Plant Safety Committee. The tasks included periodic inspection, daily measures and evaluation of the data collect at list one per year.

The following table summarizes the implementation of the AMP.

SSC	**Planned activities**
Shield	Visual inspections, revision of procedures and acceptance criteria. Assessment of results
Beam Tubes	
Building	
Biological shield	Efficiency of shielding of gamma and neutron dose to be tested every 5 years

06.2 Licensee’s experience of the application of AMPs for concrete structures

TRIGA MARK II

The Licensees acknowledge that the integrity of concrete containment should be guarantee in any condition. The application of the AMP is related to perform the ordinary maintenance and periodically re-evaluate new requirement.

06.3 Regulator’s assessment and conclusions on ageing management of concrete structures

With regard to TRIGA-RC1 reactor ISPRA has taken note of the Licensee programme to reassess concrete structures seismic resistance. A specific regulatory oversight activity is envisaged.

With regard to TRIGA MARK II it is recognized that several activities have been carried out in the recent times in order to monitor and identify ageing effects on concrete containment. A specific ageing management programme is also implemented.

07

Overall assessment and general conclusions

Italian Licensees of research reactors have been requested to submit an updated evaluation of safety which takes into account the recommendation of the Code of Conduct. This periodic review is foreseen in the context of the report on the Status of conservation of the plant and operating experience that has to be submitted every 5 years as a condition attached to the licence.

Taking into account the age of the reactors particular attention is devoted from the regulatory point of view to the proper management by the licensees of ageing mechanism, either related to physical ageing, leading to the degradation of the physical characteristics of SSCs, or to non-physical ageing related to technological obsolescence.

An evaluation of aging assessment has been requested to Licensees in connection with the Topical Peer Review 2017.

Aging management has been implemented along the years as result of the application of the surveillance rules associated to conditions and technical specifications attached to the license. This has lead to the implementation of several upgrading measures.

As far as concern TRIGA Mark II, it is recognized that in the recent times an Integrated Management System is implemented in accordance with International Standard ISO 9001. On that basis an ageing management programme (AMP) is also in place. The outcome of the comparison of the IAEA guideline SSG-10 “Ageing Management for Research Reactors” with the IMS in place leaded to the drafting of a first a road map for all the activities not already included or providing adjustments to incorporate ageing in the main management system.

The proper application of this AMP is monitored by ISPRA.

With regard to TRIGA RC-1, ISPRA has taken note of the Licensee plan to implement in 2018 upgrading modifications on infrastructures of the reactor building, on the electrical power supply and on the instrumentation and control. For this purpose the reactor will be put in temporary shutdown conditions. An Ageing Management Programme to be applied after that the above mentioned interventions has been prepared.

Application documents related to the above activities will be submitted soon.

ISPRA has taken the decision to request the Licensee the conduct of the first inspection activities identified in the Ageing Management Programme for Electrical Cables, Hydraulic Circuit and Container in connection with the envisaged prolonged shutdown phase in 2018 finalized to implement planned extraordinary management interventions and upgrading measures.

Annex 1 Ageing Mechanism – Coding according to IAEA SSG10

1. Change of properties (physical, chemical, mechanical) due to neutron irradiation
2. Change of properties (physical, chemical, mechanical) of materials due to operating temperatures
3. Mechanical stresses or cracks due to temperature/or operating pressure
4. Fatigue phenomena, material consumption due to thermal cycles, mechanical load, flow, induced vibrations
5. Corrosion
6. Chemical processes
7. Erosion
8. Technology change
9. Regulatory changes, regulations, technical specification s, etc.
10. Documentation obsolescence

Replacement difficulty

A = high

B= medium

C= low

Appendix 1 - Research Reactor Characteristics

TRIGA RC-1 Nuclear Research Reactor

General Description

TRIGA RC-1 is a pool thermal reactor with core housed inside a cylindrical graphite reflector on the bottom of an aluminium container.

The container is filled with demineralized water that has moderator function, middle-temperature refrigerant and biological shield.

The fuel consists of cylindrical elements made by a ternary alloy of Zr, H and Uranium enriched at 19.9% in U^{235} (total core about 4 kg U^{235}).

Moderation is entrusted not only to the cooling water but also to the alloy zirconium hydrate which is responsible for the high temperature negative prompt coefficient. The reactor is controlled by means of three shim rods (a safety bar, two fuel follower) and a fine regulation rod.

The fuel elements, end capped by two cylinders of graphite, are coated by a stainless steel cover.

An Americium/Beryllium source provides gradual and controlled rise to criticality.

The removal of the thermal power produced by the core is carried out by water natural convection.

Pool water is maintained at a constant temperature by means of a special cooling circuit equipped with heat exchangers and cooling towers.

Basic components

Core - It is constituted by a cylindrical set with a compact configuration that includes fuel elements, control bars and the Americium/Beryllium source. The water surrounding these components occupies about 1/3 of the volume of the core. The upper and lower graphite end caps of the 111 fuel elements (as of December 30, 1998) act as upper and lower reflectors of the core.

The core elements are housed between two aluminum grids, which are surrounded and supported by the reflector structure. The top grid has 127 holes for fuel elements, control bars, source, pneumatic transfer tube, and a central vertical channel for samples irradiation.

Reflector - The reflector surrounding the core consists mainly of a cylindrical graphite ring protected with an anodized aluminum outer coating water-resistant. This ring presents four horizontal penetrationsfor the radial channels A, B, C and the tangential channel used for the irradiation experiments. On the outer side of the graphite ring, a thermal lead shieldis applied inside the aluminium outer shell by means of a molybdenum coating: it is designed to improve the heat conduction to the pool water. An annular lead ring has been implemented to reduce the thermal gradients in the biological shield due to gamma rays absorption. The outer part of the reflector is holed only in correspondence with the tangential piercing channel and ends with two connecting flanges connected by bellows in order to guarantee continuousness with the parts of the channels passing through the biological concrete shield.

Fuel elements - The ternary alloy contains 8.5% by weight of Uranium, enriched to 19.9% in U^{235}. The average weight of a fuel element is about 3.4 kg, with an average content of U^{235} of about 38 g. The stability requirements of the ternary alloy grid impose a metallurgical condition resulting in a maximum variation of the number of atoms of the alloy of 1%. Obviously, in the fuel element, due to consumption, there is a reduction in the number of U^{235} atoms. For this reactor, the percentages of the ternary alloy components are established in such a way that, in spite of the total theoretical consumption of the entire U^{235}content, the variation in the number of atoms in the alloy is about 0.7%.

The fundamental feature of these fuel elements, as has been said, is represented by the temperature negative prompt coefficient, which gives the reactor a high degree of intrinsic safety: if a power escalation occurs, the reactor would automatically return to normal values.

Control rods - The reactor power is controlled by means of a fine adjustment bar without "fueled-follower" and three "fueled-follower" adjustment bars (with "following" fuel element).

The fine adjustment bar is constituted by an absorbent element of aluminum-coated boron carbide. The three "fueled-follower" bars (two coarse, "shim", and one "safety"), comprise a portion of borated graphite absorber, which has in the lower part a fuel element..

The bars are coated by stainless steel pipes, identical to those of the fuel elements.

Control System - Nuclear power measurement instrumentation includes: a starting channel, two extended range linear channels, and a safety channel.

Radiation monitoring equipment includes several devices that measure the activity of refrigerant and air, as well as environmental radioactivity.

A number of conventional variables (temperatures, pressures, flow rates, etc.) are also measured with the aim of providing an accurate picture of the working conditions of the plant.

All linear power measurement channels, radiation monitoring instruments, and part of the instrumentation for conventional variables measurements, include protection devices designed to provide alarm signals and, where appropriate, automatic emergency control rods operation.

The starting channel measures and controls the neutron flux from the "source" field beyond the power corresponding to criticality. It includes a fission counter followed by a preamplifier, a discriminator, a logarithmic impulse meter, and a linear variable sensitivity.

Extended range linear channels measure and control the neutron flux in a power range between 10^{-2} W and 1 MW. Each of them includes a compensated ionization chamber, a linear amplifier followed by two threshold circuits designed to provide alarm signals and automatic control bar operation.

The safety channel is active in the last two decades of the power range (10 kW and 1 MW). It is constituted by an uncompensated ionization chamber followed by a threshold amplifier with fast response time. The instrument operated by the amplifier indicates how much actual power differs in percentage from the trigger level fixed by a handhold.

Table A1 – TRIGA RC-1 - Main characteristics of the reactor

Power:	1 MW
Max Thermal Flux:	$2.7 \cdot 10^{13}$ n/cm^{2}s
Fuel:	max number of F.E.:140 composition: U–ZrH weight U^{235}: 38 gr contents in weight of U: 8.5% enrichment: 19.9% cladding: AISI 304
Control rods:	- 3 boron/carbide “fuel follower” (shim, safety) - 1 boron/carbide (regolating) - max. reactivity margin: 4500 pcm - min. shut down margin: 500 pcm - max. Thermal jump in the core: 35 °C
Irradiation facilities:	- 5 radial channels - 1 central thimble (vertical) - 1 lazy susan (40 samples) - 1 rabbit (vertical) - 1 thermal column (horizontal) - 1 thermalizing column (horizontal, in the shielding tank)

TRIGA Mark II Nuclear Research Reactor

Since 1965, at the University of Pavia the Applied Nuclear Energy Laboratory (LENA), operates a TRIGA Mark II reactor developed by General Atomics (the General Research Atomics Training Research Isotopes). This reactor named TRIGA MKII, operates at a power of 250 kW.

Type of reactor

The TRIGA Mark II is a multipurpose Reactor that can be used on advance materials studies, radioisotope productions and analysis by nuclear techniques, whether for educational and training.

The basic TRIGA reactor design is an open pool configuration cooled by natural convection flow through the core with demineralized water. The standard TRIGA fuel elements contain 8.5–12 wt% of nominally up to 20% enriched uranium using a homogenously mixed moderator designed on the principle of inherent safety for reactivity insertion events utilizing the UZrHx alloy. The reactor core and the reflector are placed on the bottom of an aluminium container of 1.98 m in diameter and 6.2 m in depth. About 5.0 m of water above the core are the vertical shield. The core is then shielded radially by 30.5 cm of graphite of the reflector, 45.7 cm of water and a minimum of 2.3 m of ordinary cement (2.3 g/cm^3 density) of the biological shield. The core is a fixed cylindrical geometry, approximately 0.58 m × 0.45 m diameter surrounded by a graphite reflector. The core consisted of a lattice of fuel moderator elements, graphite dummy elements, and three control rods located within element positions. The two grid plates held the fuel moderator elements in a vertical position and allowed cooling water flow to flow by natural convection around the fuel elements to remove the heat generated from fission. In addition to the fuel element positions there are a central chamber (38.4 mm diameter) located at the center of the core and sixteen foil-insertion holes (8 mm diameter) inside the reactor core configuration. The central chamber is used for irradiating larger sample specimens using an irradiation thimble.

Fuel elements are equipped with two 10.19 cm high graphite cylinders, located at the two lower and upper extremities, which function as reflectors. A 30.5cm radial graphite reflector surrounds the entire core and is place on an aluminium stand on the bottom of the container. Water take up about a third of the volume of the core.

The experimental and radiant irradiation equipment is remarkable and versatile. It is possible at any time to access and observe the core through the vertical water shield. A graphite column extends from the outside of the reflector inside the biological shield (north side). Since 1992, some of the graphite has been replaced by bismuth and a radiant cavity has been formed, characterized by a highly thermalized neutron spectrum.

Horizontal access and shielding of the thermal column is provided by a barite-laden concrete door (about 3.5 g/cm^3 density) running on rails.

In 1978 an external thermal shielding structure was constructed with two concrete shutter slides transversely to the door; The alternate opening of the shutter and the door allows you to place samples to be irradiated in the column even with the reactor in operation (in case of simultaneous opening the reactor is automatically stopped by a safety circuit).

A column of graphite (called thermalisation column) smaller than the thermal column extends from the opposite face of the reflector (south side) within the biological screen.

A 2,66-meter-long, 2,44-meter-wide, and 3,88-meter deep pool (pool) serves as a shield and access to the workplace thermalisation column. Four irradiation channels extend from the reflector through the water and the concrete to the outside of the shielding structure.

A rotating specimen holder, located in an annular seat at the top of the radial graphite reflector, is used for the production of radioisotopes and the irradiation of small samples. The TRIGA Rector possesses a Central Channel for experiments, positioned in the center of the core pattern, that is at the point of maximum neutron flow.

Vertical canals for experiments can easily be installed in the core region to provide additional irradiation systems for samples with high neutron flows. A pneumatic transfer system allows the insertion of samples into the core (ring F) and the production of short-lived radioisotopes. An aluminium vertical channel (Channel F) can be positioned in the same ring F for sample irradiation.

The power level of the TRIGA is controlled by three control rods. Pulsed operation tests have proven that intrinsic safety of the TRIGA reactor offers effective self-regulation of power without the need for automatic control and shutdown systems.

Three measurement channels provide information on the neutron flow and reactor power level and period.

The reactor's primary circuit water is monitored by means of instrumentation measuring temperature, conductivity and radioactivity. An ion exchange resin system provides continuous water purification of the primary circuit.

Instrumentation

Reactor instrumentation includes three neutron measurement channels, radioactivity, temperature and water conductivity monitors, a fuel temperature measurement circuit, one for peak neutron flow measurement, and a "console" of control. The neutron channels consist of three neutron-sensitive ionisation hermetically sealed in aluminium boxes and mounted around the reflector with a support structure. One of these, for the low stairs, includes a fission counter. The type of chamber used and its location determine the neutron flow measurement range.

The primary hydraulic system

The primary hydraulic system serves the following four functions:

1. It maintains low water conductivity to minimize corrosion of all reactor components, in particular fuel elements;
2. Reduces water's radioactivity by removing almost all solid particles and soluble impurities;
3. Keeps the transparency of the water;
4. Provides a means for subtracting the heat generated in the reactor by subtracting it to the water from the secondary cooling system by means of a water-to-water heat exchanger.

The primary system applied to the 250 kW TRIGA Mark II Reactor consists essentially of a pump, a fiber cartridge filter, and a heat exchanger connected to appropriate tubing and aluminium valves.

A surface foam helps to keep the surface of the reactor container water clear; A monitoring vessel through which the primary circuit water flows, contains probes for temperature, radioactivity and water conductivity measurements

Secondary cooling hydraulic system

The secondary cooling hydraulic system serves to subtract the heat from the primary hydraulic system and disperse it appropriately. In order to prevent any possibility of pollution of the drained water in the city drainage, the secondary hydraulic system consists of two distinct circuits separated by a second heat exchanger: a closed circuit, which constitutes the proper secondary circuit and an open circuit Tertiary) with drainage.

Table A2 – TRIGA MARK II - Flows in the various experimental areas

Position	Flux		
	neutron/cm^2-sec		Gamma (rads/sec)
	fast (>10 Kev)	thermal (< 0,21 Kev)	
Central thimble	$9x10^{12}$	$9x10^{12}$	$2,5x10^{4}$
Ring "E"	$6,4x10^{12}$	$4,1x10^{12}$	$1,5x10^{4}$
Pneumatic system (ring “F”)	$4x10^{12}$	$4x10^{12}$	$1,5x10^{4}$
Piercing channel	$2,0x10^{12}$	$2,0x10^{12}$	$1,0x10^{4}$
Rotary specimen	$1x10^{12}$	$1,2x10^{12}$	$1,0x10^{3}$
Reflector zone	$6,8x10^{10}$	$6,8x10^{11}$	$4,5x10^{2}$
Water zone over the reflector	$1,1x10^{11}$	$3,4x10^{11}$	===

Appendix 2 - Characteristics of the electrical system of the TRIGA RC-1

The electrical system of the TRIGA RC-1, both for the reactor and the offices area, is powered by a low voltage line connected to an electrical substation located near the plant fence.

In the substation there are two transformation groups of 1000 kVA each and a converter unit powered by a series of 290 Ni-Cd type units. Transformation groups, both powered by voltage at 8400 V, return to the secondary voltage at 400/220 V.

In case of a power failure, there are two emergency systems: the mentioned converter unit and an electro-diesel unit for the supply of the Casaccia Research Center essential services.

The converter unit only supplies the following services:

1) Reactor and Heath Physics Instruments;

2) Auxiliary Reactor Console;

3) Reactor console;

4) Shutters;

5) Reactor shaft light;

6) Reactor Room Lights.

The electro-diesel unit also supplies, in addition to the above mentioned, the following components:

7) Cooling tower 1;

8) Cooling tower 2;

9) Primary pump 1;

10) Secondary pump 1;

11) Reactor Counters (Argon Drain Pump);

12) Thermal column door engine;

13) Primary Pump 2;

14) Secondary pump 2;

15) Derived pump 1;

16) Derived pump 2;

17) Primary pump 3;

18) Secondary pump 3;

19) Bridge crane;

20) Ventilation system.

The converter unit can supply a power of 60 kVA for a time of about 20 minutes and is basically constituted by a battery and an inverter group that returns alternate current.

The nickel-cadmium accumulator battery consists of 290 units of 1.4 V each. The available voltages at the output of the inverter are 380/400/415 V with currents of 91/87/84 A respectively. The battery is normally held under charge via a rectifier powered by the electrical grid.

The converter unit is equipped with a control panel on which all the systems needed for the correct and safe operation of the unit itself are mounted and connected (measuring, regulation, control, protection and alarm instruments).

The users reported from 1) to 6) are directly connected to the converter unit, because for them it is not possible to allow the dead time due to the automatic insertion of the Electro-diesel Group (about the first one minute after the failure of electrical supply) without interrupting the operation of the reactor. This means that the converter unit delivers current only for one minute, using only a fraction of the battery life.

From the external transformer station, the voltage is distributed to a series of electrical panels that supply the plant and offices. The main panel is located in Control Room and supplies:

1) Control Room Panels;

2) Argon Drainage;

3) Experiment Pool Pump Circuit;

4) Drainage Hall Pumps;

5) Calibration 1;

6) Calibration 2;

7) Refrigeration Tower No.1;

8) Refrigeration Tower No. 2;

9) Primary Circuit Pump No.1;

10) Primary Circuit Pump N.2;

11) Primary Circuit Pump N.3;

12) Secondary Circuit Pump No.1;

13) Secondary Circuit Pump N.2;

14) Secondary Circuit Pump No.3;

15) Reactor Room Panels;

16) Thermal Column door;

17) Bridge Crane;

18) Reactor Panels and Health Physics Instruments;

19) UPS and alarm control;

20) Main and Auxiliary Console;

21) Shutters;

22) Reactor Room Panels and Illumination;

23) Reactor Room Illumination N.1;

24) Reactor Room Illumination N.2;

25) Emergency Illumination;

26) Reactor Well Illumination.

The other Low Voltage Panels are:

1) Basement;

2) Ground Floor;

3) First Floor;

4) Thermal Power Station;

5) Refrigeration Central;

6) Central ventilation plant;

7) Illumination and Engine Center;

8) Engine Center Normal operation;

9) Engine Center Emergency;

10) Control Room 110 V DC.

The power substation was updated in 2001 and the inverter replaced in 2011. Both fulfill all actual law requirements.

During the upgrading process of the substation, the power supply cables for low voltage electrical panels and the internal distribution network for office areas were also replaced.