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Development of the Reliability Assurance Program in a Brazilian nuclear power plant subsidized by a Reliability, Availability and Maintainability Model

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ABSTRACT

The main objective of this work is to present a methodology for the development of a Reliability Assurance Program (RAP) specific to a PWR experimental nuclear installation, through the analysis of the installation and the development of a preliminary RAP subsidized by a Reliability, Availability and Maintainability (RAM) model. The study of an evaluation was carried out in the long-term decay heat removal of the studied experimental plant, whose data were used for application of the RAP. The necessary steps for applying the developed RAP are followed, using the data from the assessment of the studied plant, resulting in a list of components of significant risk for the Program, and in the following steps of sending the list to the experts panel, ranking of SSCs by the panel and development of the final list of significant risk SSCs for using the list in the optimization of the plant. The RAP subsidized by a RAM model will be able to work with the logical relationships between each component of the plant for their effects on energy generation and with the quantitative prediction of the magnitude of each contributor to the occurrence of high-level events, and the developed methodology can be applicable throughout the experimental plant. In this way, it will be possible to implement the RAP in the plant, which will provide a structured way to meet the regulatory requirements for its licensing. Also, it will be possible to complement the plant safety analysis report, which must contain the RAP.

Keywords: **Reliability Assurance Program, Brazilian nuclear power plant, Reliability Availability and Maintainability.**

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1. INTRODUCTION

In 1996, the International Atomic Energy Agency (IAEA) began the task of developing a reliability assurance guide to support the implementation of advanced reactor programs and facilitate the next generation of commercial nuclear reactors to achieve a high level of safety, reliability and economy [1]. A RAP must be part of the implementation of nuclear installation management programs, so that these reach a high level of safety, reliability, quality and economic performance.

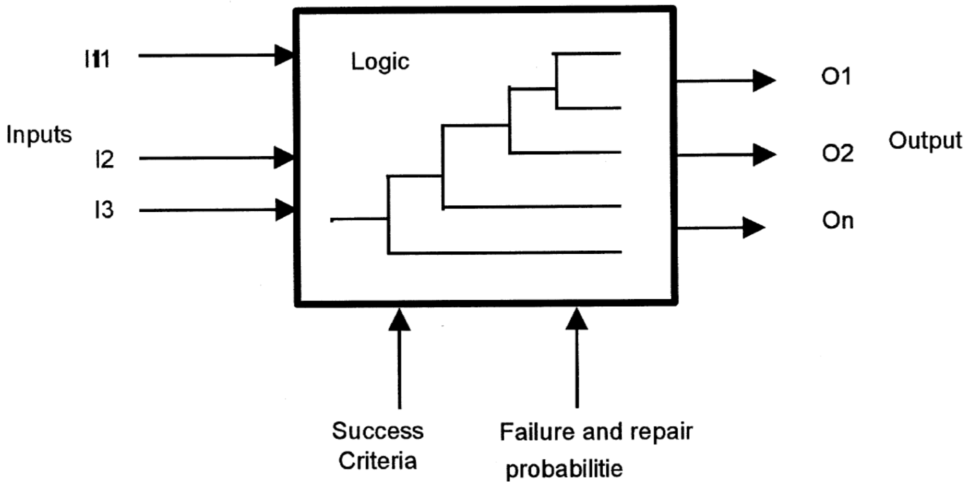
Techniques and methods normally used in Reliability Engineering and Risk Analysis can be applied to increase the training of nuclear reactor designers, builders and operators, with a view to developing a specific reliability assurance program for a nuclear facility. The guide developed by the IAEA [1], TECDOC-1264, is intended for commercial nuclear power generation plants. It is also intended to augment, not replace, the specific reliability assurance requirements defined by plant requirements documents and by individual nuclear steam supply system (NSSS) designers.

The RAP applies to systems, structures and components (SSCs) significant to nuclear power plants, as determined using probabilistic, deterministic or other methods of analysis, including information obtained from sources such as industry operating experience, failure databases of relevant components and expert panels [2]. It is important to note that RAP SSCs must not degrade to an unacceptable level of reliability, availability or condition during plant operations [3]. An overview of the interactions between each functional element of the RAP, typical plant design, and operational information systems are shown diagrammatically in Figure 1 and Figure 2.

According to the IAEA [1], the main safety goals are usually defined by the regulatory authority in the form of a set of requirements that focus on critical safety issues:

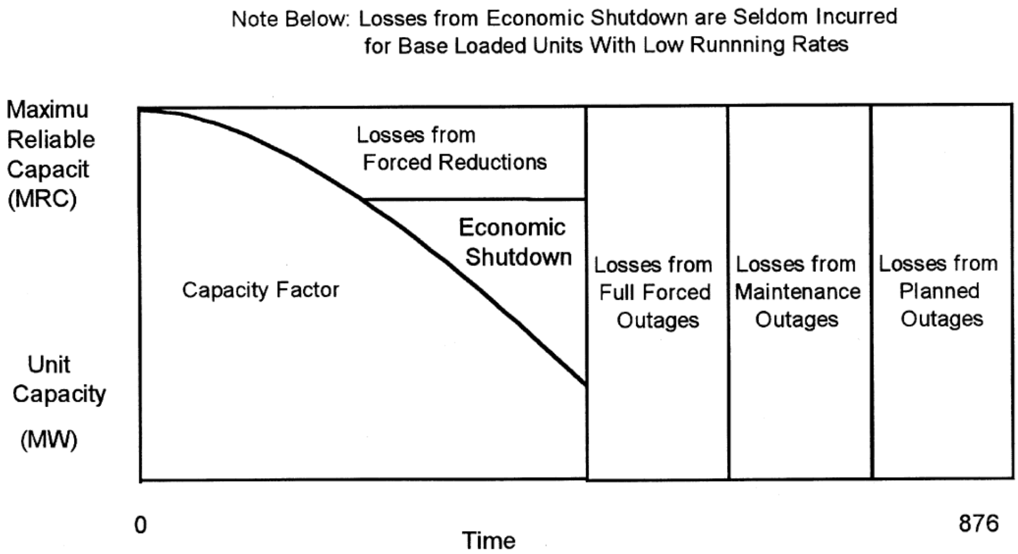
a) limit accidents by setting an acceptably low rate of occurrence measured as core damage frequency (CDF);
b) limit the release of hazardous levels of radioisotopes by setting an acceptably low rate of occurrence for a large release measured as the frequency for a large early release frequency (LERF);
c) limit harm to the general public by specifying acceptably low levels of individual risk from both the chronic and acute effects of radiation.

Figure 1: *Example of a functional model*



Source: TECDOC-1264 [1]

Figure 2: *Unit load duration curve*



Source: TECDOC-1264 [1]

The object of study of this work is to develop a preliminary reliability assurance program (RAP) in a Brazilian nuclear plant supported by a Reliability, Availability and Maintainability (RAM) model. The implementation of a RAP is justified by the fact that it provides a structured way to meet the regulatory requirements of nuclear plants. The RAP itself is already a regulatory requirement, which must be present in the plant's Safety Analysis Report [4]. It should be noted that the methodology presented in this work is an adaptation, for an experimental installation, of existing practices focused on conventional energy production plants. Thus, the peculiarities of this adaptation will be discussed, and the objectives of the experimental plant will be used as criteria for reliability optimization.

The general objective of this work is to present a methodology for the development of a specific RAP for an experimental PWR nuclear installation being implemented in Brazil.

Therefore, the specific objectives will be:

a) analyze the PWR experimental installation being implemented in Brazil;

b) develop a preliminary RAP supported by a RAM model

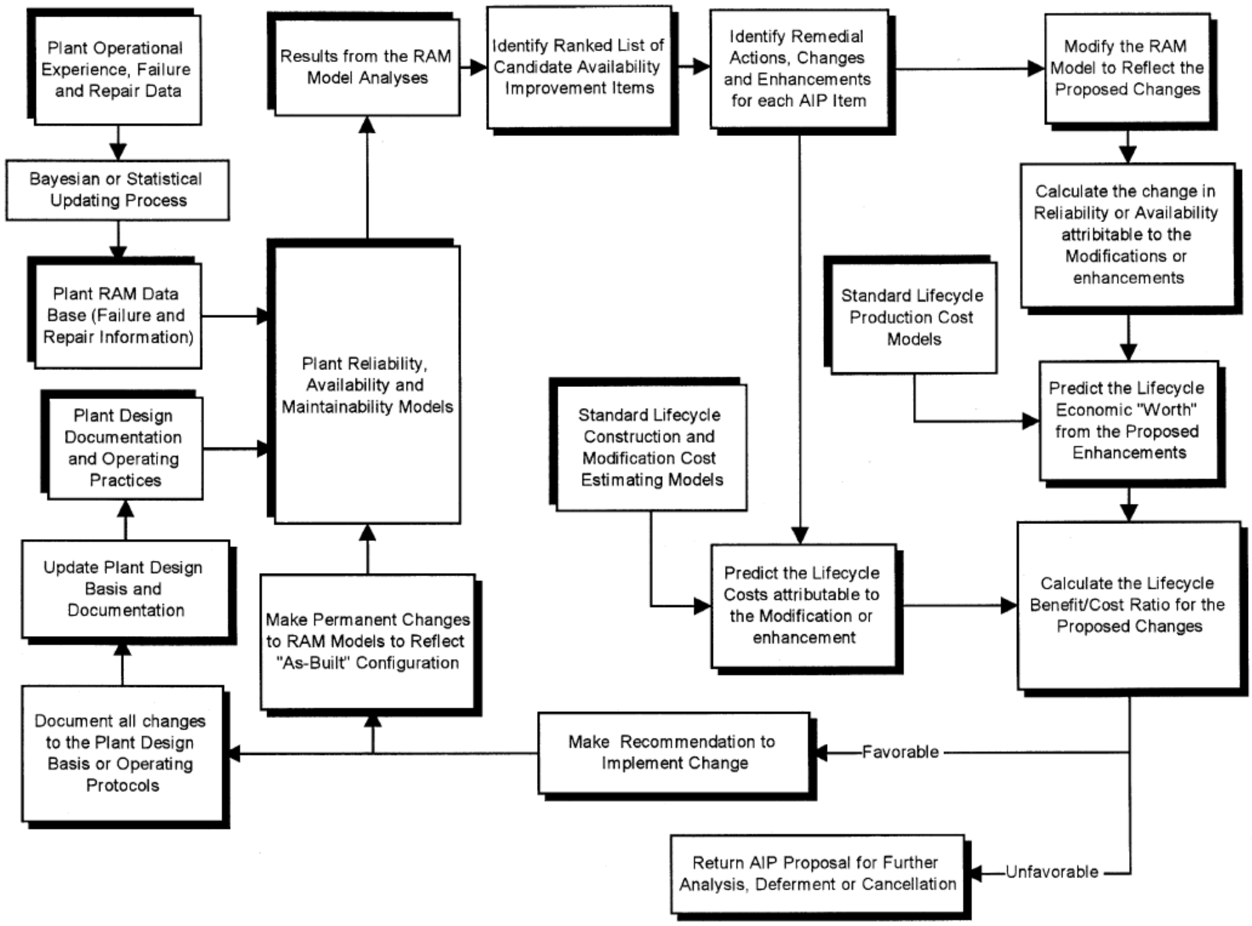
In order to achieve these objectives, it was necessary to carry out extensive bibliographical research in the databases of the core area available in order to understand the various aspects that involve the RAP as a formal management system, such as the techniques and programs available for its application.

2. MATERIALS AND METHODS

The methodology elaborated for the development of the RAP is based on the TECDOC-1264 guide of the IAEA [1], on the requirements of item 17.4 of the NUREG-0800 of the U.S. Nuclear Regulatory Commission [4] (NRC) and in the researched bibliography, considering the objectives and needs of the experimental nuclear plant object of this work.

According to the cited guide, a formal RAP will fully integrate plant management systems and associated activities that influence the reliability, availability or maintainability of critical SSCs whose performance is important to plant capacity factor, plant safety or plant risk. A process that can be used to incorporate the use of RAM models in RAP, is shown in Figure 3. It should be noted that there is a conflict between cost and safety when preparing the RAP. Equipment with a high (maintenance) cost and little impact on safety can be considered less rigorously in the RAP.

Figure 3: *RAM models in reliability assurance programs*



AIP: availability improvement programme

Source: TECDOC-1264 [1]

Therefore, the proposed RAP will apply to SSCs that are identified as being of significant risk (or significant contributors to plant safety) as determined using probabilistic, deterministic and other methods of analysis, industry operational experience, databases of relevant component failures and expert panels. The implementation of the RAP will provide a power plant that operates reliably and safely while generating electricity at minimal cost and fully meeting all safety requirements [1].

To illustrate the operation and results of a RAP in the nuclear plant using the methodology presented, an analysis of an assessment carried out in the experimental plant studied in this work was executed, for the application of the RAP to identify SSCs of significant risk. This work utilizes an assessment performed on the potential risk of losing long-term decay heat removal during a refueling shutdown.

The systems working in the long-term decay heat removal are the residual heat removal system, the fuel pool cooling system, the safety water system, the primary component cooling water system, secondary fuel pool cooling system, and AC and DC electrical power. The **residual heat removal system** is a subsystem of the emergency cooling system. The residual heat removal system removes residual heat generated in the reactor core, and the proper performance of the residual heat removal system is of critical importance for safe nuclear operations at the experimental power plant.

The function of the residual heat removal system is to provide long-term residual heat removal in the reactor core after normal shutdown, a loss of coolant accident (LOCA), and transient scenarios. This subsystem is in standby during normal power operation and is immediately actuated by the prototype protection system after the normal or emergency shutdown of the reactor, if the residual heat cannot be effectively removed by the steam generators.

The residual heat removal system ranks second among 13 systems in contribution to core damage frequency (CDF). The residual heat removal system provides long-term decay heat removal following any initiating event. A failure to remove decay heat will ultimately lead to core damage. The residual heat removal system is a two-train system in which only one train is required for success. The passive heat exchanger was not credited in that study, but would add a third level of redundancy for long-term residual heat removal. The top contribution from this system involves plugging of both train 1 and train 2 heat exchangers which precludes long term cooling to the reactor building system.

The **fuel pool cooling system** is designed to remove the residual heat from the existing spent fuel elements, caused by the decay of fission products. The fuel pool cooling system is utilized during shutdown phases, when fuel is present in the spent fuel pool / transfer canal. The fuel pool cooling system is composed of two independent loops, in order to avoid the mixing of water between spent fuel pool compartments.

There are two cooling loops present for the fuel pool cooling system. The loss of fuel pool cooling system loop 2 initiating event is the top contributor to core damage during shutdown among the analyzed initiating event categories. Key component failures that lead directly to fuel damage include the failure to open fuel pool cooling system manual valves 034 and 036 and the failure of fuel pool cooling system check valve 018 to open. Other important component failures include the common cause start failures of fuel pool cooling system pump train 2A/2B and the unavailability of fuel pool cooling system pump train 2A/2B, due to test and maintenance.

The **safety water system** provides cooling water to the secondary side of the primary components cooling system and acts as the ultimate heat sink for the residual heat removal subsystem of the emergency cooling system. The safety water system is composed of three subsystems: the security cooling subsystem, the critical systems cooling subsystem, and the water replacement subsystem. The proper performance of the security cooling subsystem and the critical systems cooling subsystem is of critical importance for the proper cooling of the plant during normal power, post-trip, and accident operations.

The safety water system is the fourth highest contribution to CDF from a system. The safety water system is responsible for cooling the shielding pool and the primary component cooling system heat exchangers, via evaporative coolers. Failure to cool the shielding pool or the primary component cooling system heat exchangers will ultimately lead to a failure of the residual heat removal system, resulting in core damage.

The **primary component cooling system** transfers heat from operating equipment in the reactor building to the safety water system to ensure that the equipment is operated within design parameters. There is also an initiating event for loss of primary components cooling water. The function of the RCP system is to provide cooling to heat loads in the reactor building. This system is normally in operation with one train aligned for cooling and the other train in standby.

The operating train is aligned to provide cooling to the following components during normal operation:

d) Reactor cooling system circulation pumps (1A, 1B, 1C, 2A, 2B, and 2C)
e) Primary purification system heat exchangers (1 and 2)
f) The 21 fuel assemblies of the control rod actuation mechanism
g) Primary coolant discharge system heat exchanger (1)
h) Reactor shield water system shielding tank (2)
i) Residual heat removal system circulation pumps (5 and 6)

The primary component cooling system is responsible for supplying component cooling water to residual heat removal system pumps 5 and 6. A primary component cooling system level failure will preclude adequate cooling to the appropriate residual heat removal system pump and consequently fail that specific residual heat removal system train. A loss of both trains will lead to the inability to provide long-term cooling to the reactor during shutdown.

The **secondary fuel pool cooling system** is designed to serve as a source of water for cooling the heat exchangers from the fuel pool cooling system. The secondary fuel pool cooling system consists of a closed loop with circulating pumps (1A/1B), evaporative coolers (1A/1B), and expansion tanks (1A/1B), forming two fully redundant, interconnected trains so that each pump, expansion tank, and evaporative cooler can be shared. The secondary fuel pool cooling system always works in conjunction with the fuel pool cooling system.

Pumps 1A/1B circulate the water from evaporative coolers 1A/1B through the three fuel pool cooling system heat exchangers, where the water returns to the chillers to be cooled, closing the loop. Changes in the volume of the secondary fuel pool cooling system are accommodated by the expansion tanks. While both secondary fuel pool cooling system trains are running during shutdown, only one train is required to successfully cool the fuel pool cooling system heat exchangers.

The **ac electrical power system** is a support system that provides ac power to all safety-related systems in the experiment plant facility. There are several different subsystems that involve different voltages of electricity, depending on the load. There are two power distribution systems, called substations, each with different functions. Substation 1 provides power to all loads requiring 13.8kV ac power and also 460V ac power to all non-safety related loads. Substation 2 powers 460V ac safety buses 001 and 002 which provides 460V ac power to the lower voltage safety-related loads.

The 460V ac power system ranks fourth among 13 systems in contribution to CDF. The most significant contribution to core damage from this system is the loss of external electric power system. While the failure of external electric power system to provide power to both train A and B safety buses following a reactor trip is significant, a failure of external electric power system will not lead directly to core damage and must be coupled with other experimental plant system failures to do so. The loss of these safety buses effects the availability of safety-related loads at experimental plant and would require reliance on backup power systems.

The **120/208V ac electric power system** provides ac power to plant equipment that is necessary for all phases of plant operation, including startup, power operation, plant shutdown, and emergency plant conditions. The function of the 120/208V ac power system is to provide continuous, uninterruptible power to safety-grade plant systems that require 120/208V ac power during normal operation and emergency conditions.

The 120/208V ac power system ranks ninth among 13 systems in contribution to CDF. The most significant contribution from the 120/208V ac power system to core damage involves circuit breaker/disconnect switch failures, as well as bus faults. The dc power system provides backup to the loads in which the 120/208V ac buses support, so the contribution to core damage from 120/208V ac buses is less than the contribution from the 460V ac power system.

The **dc electrical power system** is a support system that provides direct current (125V dc, 203V dc, and 250V dc) to all safety-related systems in the experimental plant facility by interfacing with the ac power system for all phases of operation. The dc power system is responsible for providing power to solenoid-operated valves. The function of the dc power system is to provide power to non-safety and safety-grade plant systems that require dc power during normal operation and emergency conditions.

The dc power system is used to provide dc power to various components in the power plant that require dc power. A review of each individual 125V dc bus within the dc power system was performed, and none were identified that would result in a failure of equipment. The 24V dc power required for pump and valve control circuits is not a separate dc power subsystem, but is initiated via ac to dc power transformers in the distributed logic systems. Therefore, failure of a 24V dc supply would be included within the failure of an ac power bus, and a special initiating event category is not required for loss of dc power bus or the dc power system.

The shutdown assessment included plant structures, systems, and equipment that are necessary to maintain plant parameters within a safe-stable state during refueling outages. This involves the following front-line systems:

a) residual heat removal system
b) fuel pool cooling system

Additionally, the safety water system (which acts as the final heat sink), primary component cooling water system, secondary fuel pool cooling system, and AC and DC electrical power serve as secondary/support systems for the success of long-term decay heat removal.

The data development effort for the assessment used involved generating appropriate reliability data for each event in the integrated plant model. For system models, data values for postulated component failures were required. Generic data from publicly available industry sources provided the initial basis for these failure rates. The experimental plant assessment did not include plant-specific performance in the data development since the facility is not currently operable.

The system modeling for the assessment used for the application of the RAP used detailed fault tree models for the systems called in the upper logic of the event tree for shutdown operations. These fault trees, in turn, trigger support systems such as the power and cooling systems required for their operation.

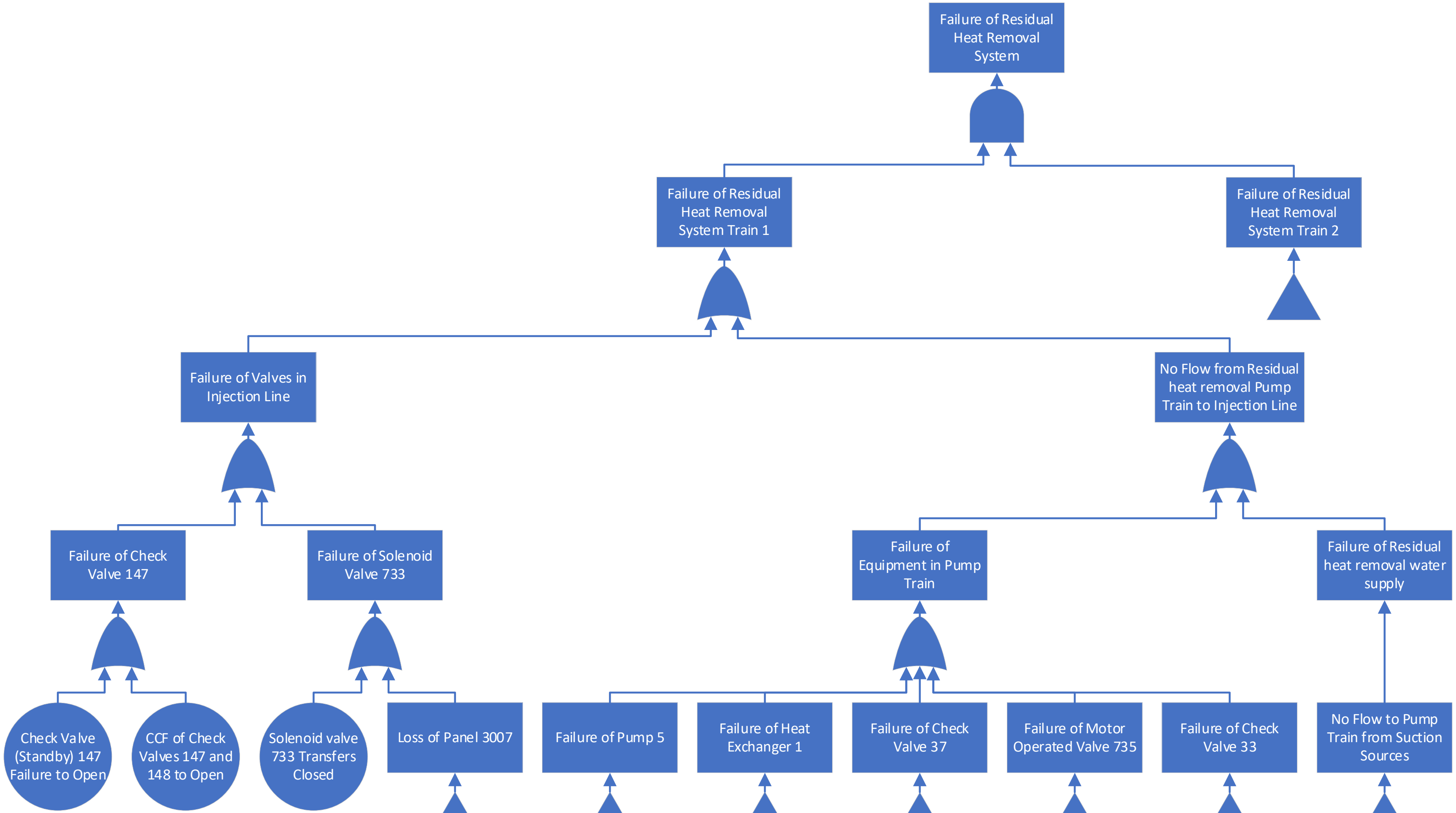
The models for this study were fully linked and explicitly included dependencies. The FaultTree+ software was used to generate, maintain, and solve the models. The models were first assembled and reviewed on a system basis to ensure correctness, then the accident sequence models were input and combined with the system level models to produce an overall linked fault tree from which cut sets are generated. Figure 4 illustrates an extract of the FaultTree+ software modeling for the plant shutdown phases, in which complete loss of the residual heat removal system includes a failure of both system trains either due to hardware faults within the system, a failure of support systems, or a combination of both. The failure of the residual heat removal system, including the impacts of support system failures, is evaluated using the linked fault tree model methodology.

The relative importance of systems and components, initiating events, etc. were calculated from the cut sets of the core damage frequency at the experimental plant shutdown. Significant risk SSCs were judged based on the risk achievement worth (RAW), common cause failure (CCF) RAW or Fussell-Vesely (FV) value of the respective SSC, calculated with the FaultTree+ software. Components with a RAW value equal to or greater than 2, CCF RAW value equal to or greater than 20, or FV equal to or greater than 0.005 were considered to be of significant risk [5].

3. RESULTS AND DISCUSSION

The plant configuration changes with time during shutdown. The fuel is not restricted to the reactor vessel (RV), and entire electrical trains of equipment can be taken out of service at a time. Factors such as these must be considered during the shutdown analysis. The boundary for the shutdown analysis commences when system pressure reaches atmospheric conditions (101.3 kPa), water temperature reaches 200°F (93.3°C), and the cold shutdown state is entered. The residual heat removal system is in operation, prior to achieving cold shutdown. The shutdown analysis includes cooldown prior to fuel movement, fuel offload, full fuel unload, fuel reload, and the preparation for the return to power.

Figure 4: *Residual Heat Removal System Fault Tree Model extract for Experimental Plant Shutdown Assessment*



The purpose of applying RAP to the systems of the long-term decay heat removal during a plant refueling shutdown was to provide assurance that the design levels of safety, reliability, availability and maintainability for all SSCs of the system meet all regulatory requirements, are proportionate to their importance to safety, reliability, risk and economics and are maintained throughout the life of the plant. The objective referring to profitability was not considered in this work; however, the issue of costs is always present in the execution of the RAP.

The system assessment for the RAP applied to identify significant risk SSCs from the long-term decay heat removal during a refueling shutdown of the experimental nuclear plant was performed following the methodology presented, according to the judgment of the specified importance measures. 393 failure modes were analyzed, culminating in the identification of 148 significant risk components. An extract of entry to the RAP List of highest FV, RAW and CCF Importance Measures is shown in tables 1, 2 and 3 respectively.

Table 1: Extract of entry to the RAP List of highest FV Importance Measures

System	Component Description	Fail mode	FV
Safety Water	Pump 1A	Unavailable due to testing or maintenance	0.1371
Safety Water	Pump 5A	Unavailable due to testing or maintenance	0.1371
Residual Heat Removal	Pumps 5 and 6	CCF to start	0.1132
Residual Heat Removal	Pumps 7 and 8	CCF to start	0.1132
Primary Component Cooling	Pumps 1A and 1B	CCF to start	0.1132
Safety Water	Pump 1B	Unavailable due to testing or maintenance	0.1044
Safety Water	Pump 5B	Unavailable due to testing or maintenance	0.1044
Residual Heat Removal	Valve 735	Failure to open	0.0860
Primary Component Cooling	B train	Unavailable due to testing or maintenance	0.0693
Primary Component Cooling	A train	Unavailable due to testing or maintenance	0.0686

Table 2: Extract of entry to the RAP List of highest RAW Importance Measures

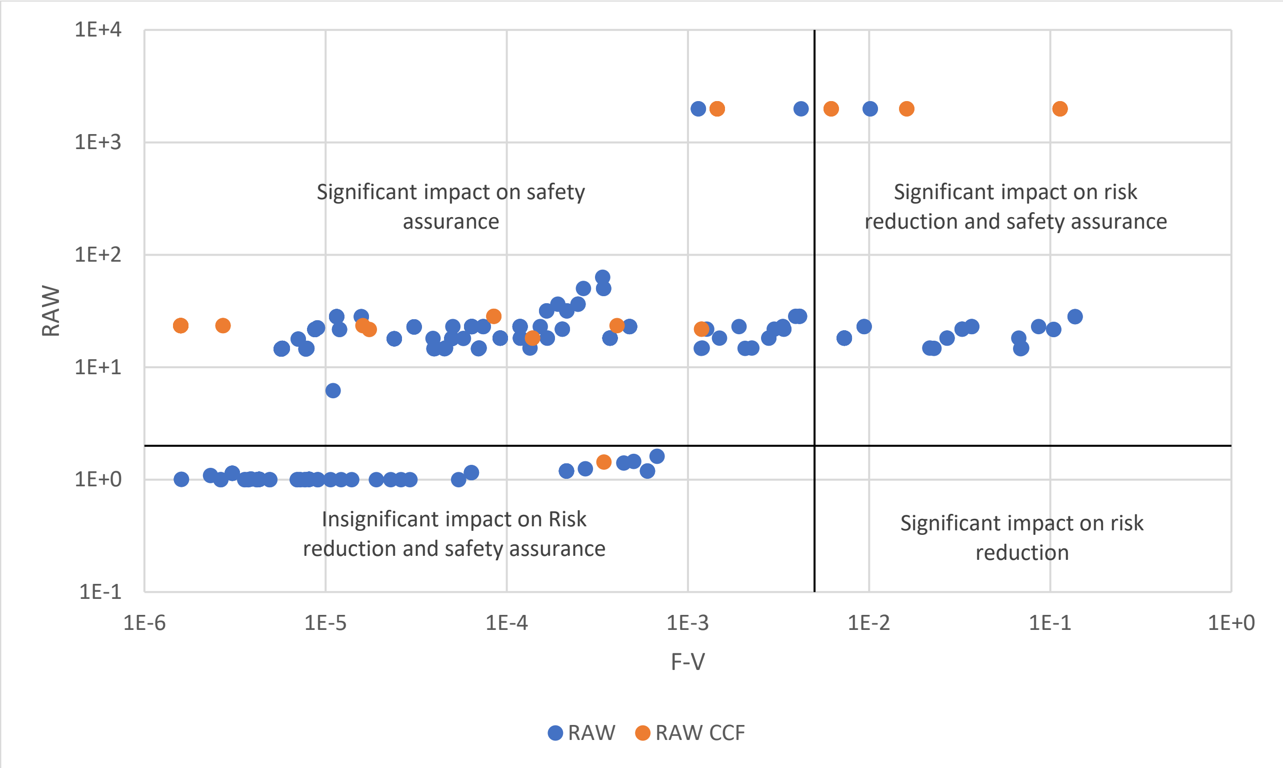
System	Component Description	Fail mode	RAW
Primary Component Cooling	Safety Valve 1594	Transfers Open	1990.12
Primary Component Cooling	Safety valve 1628	Transfers Open	1990.12
Primary Component Cooling	tank 1	Leaks	1990.12
Primary Component Cooling	Manual Valve 037	Rupture	1990.12
Primary Component Cooling	Manual Valve 064	Rupture	1990.12
AC Electric Power	Safety bus 001	Fault	63.23
AC Electric Power	460 VAC Circuit Breaker 0015	Transfers Open	49.98
AC Electric Power	disconnect switch 0130	Transfers Open	49.98
AC Electric Power	460 VAC motor control center 013	Fault	49.98
AC Electric Power	CA 0016 Circuit Breaker	Transfers Open	36.44

Table 3: Extract of entry to the RAP List of highest CCF RAW Importance Measures

System	Component Description	Fail mode	CCF RAW
Residual Heat Removal	Pumps 5 and 6	CCF to start	1990.12
Residual Heat Removal	Pumps 7 and 8	CCF to start	1990.12
Primary Component Cooling	Pumps 1A and 1B	CCF to start	1990.12
Residual Heat Removal	Heat exchangers 1 and 2	CCF Plugged	1990.12
Primary Component Cooling	Heat exchangers 1A and 1B	CCF Plugged	1990.12
Residual Heat Removal	Pumps 5 and 6	CCF to Run	1990.12
Residual Heat Removal	Pumps 7 and 8	CCF to Run	1990.12
Primary Component Cooling	Pumps 1A and 1B	CCF to Run	1990.12
Residual Heat Removal	Check valves 147 and 148	CCF to Open	1990.12
Residual Heat Removal	Check valves 166 and 167	CCF to Open	1990.12
Residual Heat Removal	Check valves 33 and 34	CCF to Open	1990.12
Residual Heat Removal	Check valves 37 and 38	CCF to Open	1990.12
Residual Heat Removal	Check valves 41 and 40	CCF to Open	1990.12

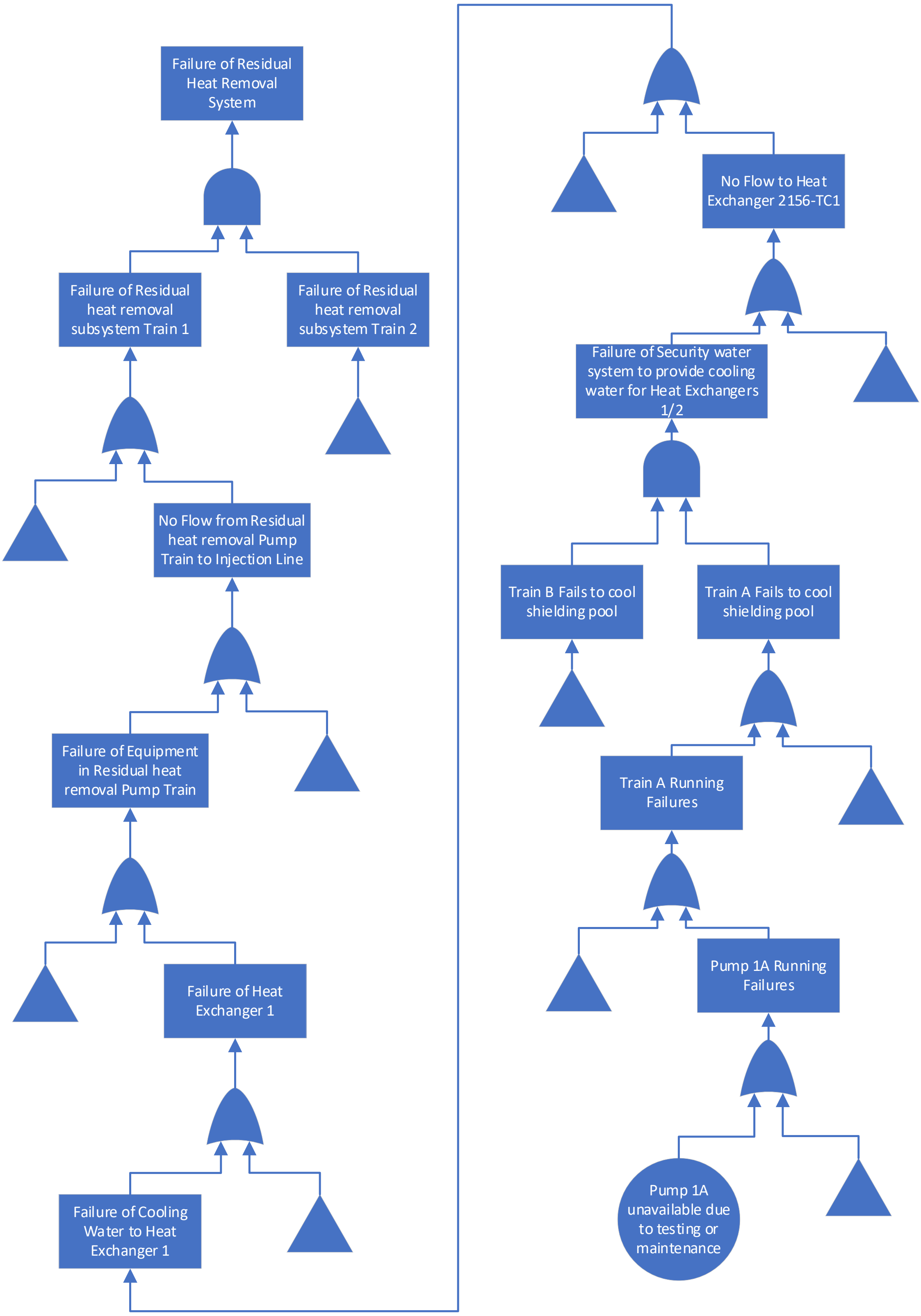
The graph showing the entry of the evaluation in the RAP applied to identify significant risk SSCs following the approach suggested by Vesely and Davis [6] is presented in Figure 5. The actual boundaries were suggested by Nuclear Energy Institute [7]. It can be noted in this graph a certain aggregation of results in three groups of values of RAW: an upper one (with values 1990.12), one with values between 10 and 100 and a lower one with values between 1 and 2. There is also a slight tendency for the RAW values to increase as the FV values increase.

Figure 5: *Using measures of importance to categorize SSCs*



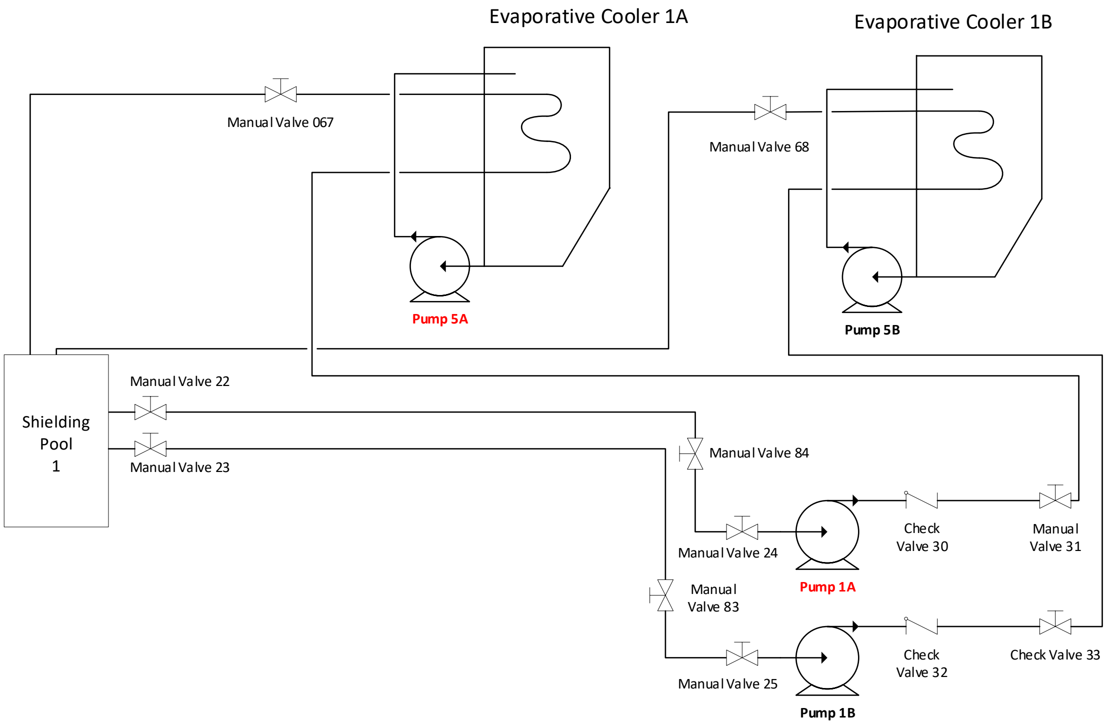
The components with the highest measures of Fussell-Vesely (FV) importance, that is, with the highest fractional contributions to the risk, are pumps 1A and 5A of the safety water system, in a situation of unavailability due to tests or maintenance, with values of 0.1371. To illustrate this situation, Figure 6 demonstrates in an extract of the fault tree how pump 1A contributes to the top event. In this example, the basic event of pump 1A unavailable due to testing or maintenance is shown in the lower right circle in the figure. Pump 1A unavailable or other base events identified by the transfer gate (the triangle to the right of the circle) lead to pump 1A malfunctions (identified by the intermediate event shown in the rectangle above the gate OR just above the base event). Those other events identified by the triangle, such as pump failure (running), CCF of pumps 1A and 1B when running, or power failure at 460 VAC motor control center 013 (as well as other events identified by the other triangles in figure), are not present in this extract of the tree shown in the figure, but were included in the study. The sequence of events shown in the figure results in the occurrence of the top event of this fault tree, that is, the failure of the residual heat removal system.

Figure 6: Pump 1A unavailable due to testing or maintenance contribution to risk



To facilitate understanding of the importance of the equipment, a representative drawing of part of the Safety water system (Security Cooling Subsystem), as discussed in the evaluation, is shown in Figure 7. This figure shows pumps 1A and 5A and pumps 1B and 5B, which are among the components with the highest FV.

Figure 7: *Safety water system (Security Cooling Subsystem) Assessment System Boundary Representative Diagram*



The next step of the RAP, according to the methodology presented in this work, would be to send the list of SSCs presented in Table 1 to the panel of experts established for the proposed RAP, with the objective of evaluating the qualitative and quantitative inputs related to the SSCs of significant risk. The expert panel would then use their experience and insights from the RAM model used to rank the RAP's SSCs, to develop the final list of significant risk SSCs.

Following the guidelines developed, a study should be carried out on the SSCs in order to improve their performance. This would be done through reductions in the failure rate and average downtime, seeking to understand and identify the causes of unavailability and including the examination of reliability and maintainability options. However, the primary focus must be determined on a case specific basis [1].

The study of the change in the performance of the SSCs will then be used to modify the RAM model to reflect the proposed changes. These changes will be used to carry out the cost/benefit evaluation proposed which, if favorable, the implementation of the proposed changes should be carried out.

4. CONCLUSIONS

The methodology of the preliminary RAP, which could be developed in this work through the extensive bibliographical research carried out, establishes the necessary steps for its application, which can be followed, since it was possible to do it with the data of the studied evaluation referring to the removal of long-term decay heat of the experimental plant analyzed in this work, which were sufficient for entry into the RAP. The RAP subsidized by a RAM model will be able to work with the logical relationships between each component of the plant for their effects on energy generation and with the quantitative prediction of the magnitude of each contributor to the occurrence of high-level events.

This approach makes possible the full application of RAP in the experimental plant, considering all the SSCs that compose it. The RAP implementation in the plant will provide a structured way to meet the regulatory requirements, such as the Brazilian norm NE 1.04 from CNEN [8], for its licensing, while it will complement the plant safety analysis report, which should contain the RAP, and achieve a high level of safety, reliability and economy.

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