CANDU-6 NPP LIFE EXTENSION ASPECTS

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The objectives of this paper is to give a highlight of the major aspects associated to plant life management and to provide as much details as it takes about the activities required to accomplish an hypothetical project of life extension for a nuclear installation, beyond its licensed design life.

The US-NRC License Renewal Application methodology is applied to a CANDU-600 type of NPP.

Keywords: degradation, life extension, critical components.

1. Introduction

This project intends to analyse the main requirements, strategies and methodologies related to NPP life extension, in particular adopted for a nuclear power plant type CANDU-600 with PHWR reactor.

The study is elaborated based on the basic concepts in this domain developed around the world, as well as on the available documentation and information related to other connective studies and ongoing researches as parts of plant life management approaches.

Particularly, for the purposes of this project, the US-NRC License Renewal Application methodology is applied to a CANDU-600 type of NPP, considering that's a systematic and detailed approach of the entire spectrum of degradation effects at structures, systems and components level [1, 2, 3, 4, 12].

The objectives of this paper is to give a highlight of the major aspects associated to plant life management and to provide as much details as it takes about the activities required to accomplish an hypothetical project of life extension for a nuclear installation, beyond its licensed design life. RAAN-SITON, in partnership with Bucharest Politechnical University, try to perform, for the first time in Romania, such a complex evaluation using the results of previous research and development activities on NPP components aging management and referenced documentation about life extension (e.g., IAEA, US-NRC, EC-DG/ Joint Research Centre).

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2. Description

This document summarizes the work performed till July 2007, treating the most important elements regarding the ways to evaluate in a comprehensive manner the problems related to NPP life extension (advantages and implications), establishment of the general methodology to evaluate the implications of NPP with reactors CANDU-6 life extension, selection of the CANDU-6 critical Structures, Systems and Components, establishment of the Nuclear Safety requirements and implications. Based on the established methodology (LRA), the study was developing an extensive analysis on degradation by ageing of a generic CANDU NPP. The activity itself intended to apply this methodology [1, 2], from the safety point of view, in order to identify those SSC which require AMR for life extension. As concerns, there were discussed in details the scoping and screening criteria as primary essential processes of AMR activities. The scoping and screening criteria used for a CANDU station, are as follows: nuclear safety function criterion; defence-in-depth criterion; seismic qualification criterion; environmental qualification criterion; structural design criterion; power availability criterion; analitical criteria;other criteria.

There were also identified the existing differences between PWR and PHWR projects, from qualitatively and quantitatively point of view, regarding the rules applied at international level for life extension activities [8, 9, 10, 12]. With this respect, in the earlier phases of this project, three main problems were analyzed namely: ● CANDU-6 project conceptual and general characteristics overview, including the operating regimes, the operating experience and the current licensing basis; ● the SSC's associated functions and component typical ageing degradation phenomenology description, as well as the used programs in a CANDU plant operation dealing with the life extension aspects; the SSC inventorying process and also the justifications of their scoping criteria applicable to CANDU project were included and discussed as well; ● the screening process of SSC essential (critical) for plant lifetime extension and the identification of ageing management programs, AMPs, required to sustain the prescribed activities from Plant Life Extension Project, are adequately detailed.

2.1. CANDU-6 station overview

The main design and operating characteristics of CANDU-6 station are presented and analyzed. There were included the installation's major parts (NSP&BOP), the plant buildings and structures, as well as the systems and their interdependencies. These details allow the identification of some specific or unique-of-kind positive aspects associated with the scoping and screening processes. From this point of view, it must be mentioned the significance of scoping criteria, more accurate and specific as nuclear safety is concerned, as a result of US-NRC requirements applied to CANDU project (e.g., defense-in-depth principle included in the nuclear safety functions criterion).

The current licensing documentation, CLB, and the modes and experience in CANDU operation are described with the intention to sustain and outline the main characteristics related to the plant operating regimes and to the quality of operating experience that are demanding new issues with immediate effects on consequential approaches, processes and activities of life extension. Faced to PWR methodology and technology, there were outlined new aspects regarding the degradation phenomenology of the CANDU components made of Zirconium alloy or of stainless steel masives, the environmental conditions imposed by the presence of pure D_2O or of heavy soluble Gadolinium salt, as well as the characteristics of the overall plant control and the existence of a number of components that can not be changed or maintained in an orderly manner (calandria vessel, calandria vault, MMS pipe segments in calandria vault, some in-core components of reactivity control mechanisms, steam generators). The plant operating modes along with the number of abnormal or transient regime cycles at which the installation is design to cope with, are described also.

2.2. Plant SSC scoping and screening processes

The study has explained the importance of the plant SSC's inventorying activity along with the detailed description of structures, systems and components associated functions (safety, process and passive functions). This process is completed not only for accuracy, but in order to sustain the plant functional and structural characteristics subjected to inherent ageing phenomenon. The scoping process following LRA rules is detailed according to the main features of CANDU project, based on the plant SSC inventory. The plant SSC's classification is using the associative groups structures defined on similarity of processes, or of safety characteristics, or of fabrication materials. As a consequence, it is possible to set up and to develop the associated functions concept, as a fundamental criterion of the SSC screening process.

The screening process is developed starting with the compliance with specific criteria settled for the SSC already classified. Those criteria are detailed and, if it is the case, extended upon specific domain according to the nuclear safety principles, criteria and requirements for a CANDU station. The applicability analysis of the screening criteria to a specific CANDU design represents an intermediate step in selection the SSC affected by the plant lifetime extension [5, 6, 12]. The entire screening process is detailed, justified and documented based on the available references, information and data, as such as the relevance of this activity don't be altered by lack of conciseness or ambiguity, especially where the differences between PWR and PHWR projects are obvious.

A SSC for which an AMR is required, is selected if at least one of the following criteria is met [1, 2]:

(i) The system or component is complying with the provisions of 10CFR §54.4(a)(1) for safety related, SR, component and with the objectives of the specific testing, inspection and maintenance programs;

(ii) The system or component is complying with the provisions of 10CFR §54.4(a)(2) for non-safety related, NSR, component and with the objectives of the specific testing, inspection and maintenance programs;

(iii) The system or component is complying with the provisions of 10CFR §54.4(a)(3) for non-safety related, NSR, component that has a specific function during design basis accident, DBA, and with the objectives of the specific testing, inspection and maintenance programs;

(iv) The system or component is complying with the provisions of associated functions criterion, others than those specified in (iii) above, according to 10CFR §54.4(b), and with the objectives of the specific testing, inspection and maintenance programs;

(v) The system or component is complying with the provisions of CANDU criterion regarding the plant parametric envelope preservation.

A CANDU-6 nuclear power plant is composed from about 1000 structures systems and components. All this SSC inventory can be divided into four categories namely [6, 12]:

♦ Limiting Critical SSC, are those SSC which require life assessment evaluation, LA, or, equivalent, a plan for life cycle management, LCM. This category consists of those SSC whose integrity and functional capability must be ensured during all operating period, including the shutdown state. These components have a significant nuclear safety impact because they are not replaceable, but are conditioning the entire plant lifetime period. In the category of limiting critical structures, systems and components are included the followings:

- calandria vessel;
- end shield components;
- calandria tubes;
- in-core reactivity mechanism components;
- main moderator system pipes in calandria vault section;
- calandria vault structure;
- containment and internal structures.

♦ Critical SSC: are those SSC which require life assessment evaluation, LA, or, equivalent, evaluation studies on systematic assessment of maintenance, SAM. This category consists of those SSC which are necessary during plant operation and while in shutdown state. From nuclear safety point of view, these structures, systems and components have a great and special significance as they represent the group of major critical SSC. Usually these components are difficult to replace because of personnel irradiation exposure, the prolonged operation in shutdown state and/or the high costs involved. The critical SSC group includes, but it's not limited to the followings: PHTS pipes and equipment; pressure tubes; steam generators; PHTS pumps; PHTS feeders; ECCS pipes and equipment; ESCS pipes and equipment; LZCS pipes and equipment; SDCS pipes and equipment; MMS heat exchangers and pumps; special safety systems components; structures with essential nuclear safety functions.

♦ Important structures and components in a variable number between 20 and 50 that require SAM studies, and Important Systems, also in a variable number between 40 and 60 which require a LCM plan used for the plant condition assessment, CA. For this group of SSC the preventive maintenance programs, the in-service inspection programs and the components condition surveillance programs can be implemented and effective, in order to mitigate the degradation by ageing phenomena, including these can be modified or replaced in orderly manner during the plant operation. The important SSC group for ageing management review, AMR, consists of the followings structures, systems and components: some of ESCS equipment; calandria vault cooling system components; PHTS feed pumps; turbine-generator system; service water systems pipes and equipment; BFWS pipes and equipment; heat exchangers; class III Diesel generators (stand-by and emergency); plant normal power supplies; class I battery-cells system; venting systems; D_2O vapour recovery system; instrument air system; digital computer control systems.

♦ Not-important SSC: are those SSC with no significance to plant life management program, PLiM, but that could induce a residual risk in ageing management assessments. In this category are included the remainder of plant structures, systems and components which are supporting the safety related SSC, and whom the preventive maintenance programs, the in-service inspection programs and the components condition surveillance programs are or may be applied in a scheduled and systematic manner. Depending on the state and performance parameters determined by analyses, evaluations, tests or inspections, these SSC can be changed (replaced) periodically without difficulty. This category of SSC includes the followings components: compressed air system components and compressors; compressed gas supply systems (N_2, H_2, He, CO_2, O_2); D_2O vapour recovery dryers; air exhaust system fans; HVAC systems; chilled water system; D_2O supply system; demi H_2O supply system; transformers; power and I&C cables systems.

2.3. Integrated plant assessment

The integrated plant assessment, IPA, and nuclear safety implications are aspects of great concerne in LRA methodology. This study deals with the all major aspects related to the effects of ageing degradation on plant safety, with the identification of analysis and evaluation processes and with the main activities developed through AMP and plant life extension. These groups of aspects are as follows: details about the nuclear safety principles, criteria and requirements as basic elements for CANDU plants design and licensing processes; identification and analysis of basic elements that structure the integrated plant assessment; nuclear safety requirements and implications analyses during the extended plant life; plant licensing renewal and PLiM strategy.

The main activities groups specified above, as parts of the project objective, are detailed in following topics:

- CANDU-6 station design and licensing bases;
- CANDU-6 safety principles and design criteria;
- Integrated plant status report after 30 years of operation: PLiM processes, technologies and methodologies associated to a CANDU plant; plant SSC screening; ageing assessment (SSC condition assessment; SSC life assessment); sistematic assessment of maintenance; integrated assessment of safety and process performances; development of technology surveillance program; moral degradation assessment studies; quality assurance progress; general conditions and technical aspects of plant life extension project; generation and management of wastes, toxic and dangerouse materials resulted from plant life extension activities; time limited ageing analyses, TLAA (pressurizer life time assessment; inlet header life time assessment; pressure tube life time assessment; calandria vessel life time assessment; industrial experience in ageing degradation assessment; probabilistic methods and models for degradation assessment; nuclear civil-work structures experience in degradation assessment); SSC condition evaluation after 30 years of operation.

As a plant life extension project evaluation synthesis through the process performances, it's recognized that after 30 years of operation the facility is beyond its maturity period, and the technical performances shall reflect this reality. The assessment of SSC after 30 years of operation was elaborated by analysis of each associative group namely, Buildings and Structures, Reactor and Mechanical Systems and Electrical and I&C Systems, at a systems and components level nominated as critical limiting, critical or important items for a CANDU station. However, from the group of nominated SSC, only the most important SSC from the safety point of view are detailed as is, those components whose weight in determination of plant status is of great concern for CANDU design. There were assessed the following: Plant Intake Structures and Buildings, Water Supplies and Sewage Systems; Fire Protection and Service Water systems; Reactor Building and Service Building; Main Integrated Building and EPS/SCA Buildings; Shutoff units, Mechanical Control Absorber and Adjuster Assemblies; Liquid zone control units and Liquid injection shutdown units; Calandria assembly; Ion chamber houses and Neutron flux detector units;End shield cooling system; Primary Heat Transport System; Pressure tubes; Feeders; Pressurizer; Steam generators; PHTS purification heat interchanger; Shutdown cooler; Purification cooler; Primary coolant purification system; Shutdown cooling system; Main moderator system Heat exchangers; Turbine-Generator system;F/M head, F/M auxiliary ports and F/M support and auxiliaries systems; Spent Fuel Transfer and Storage Systems; Main Steam System; Raw Service Water and Recirculated Cooling Water System; Spent Fuel Bay Venting System; Reactor Building Venting System; Local Air Cooler System; Instrument Air System; Turbine speed EHC system; Diesel Generators; Plant-grid power supplies connections system; Class I battery cells system; Power transformer; Instrument and power cables systems; Plant Digital Control Computers.

CANDU-6 plant life extension program includes essentially a set of activities which, from operation licensing renewal point of view, imply a systematic approach of the degradation phenomena for every selected SSC. As a consequence, the entire set of activities required during the project progress is tailored in a specific way in order to fit the main requirements and necessities of CANDU station. The main groups of such activities included into the plant life extension program, are as follows: maintenance activities during shutdown state; fuel channels replacement; nuclear systems refurbishment, including the steam generators; balance of plant systems refurbishment; reactor core initial refuelling with fresh fuel; reactor start-up and power increase, including maintenance activities during extended period of operation.

2.4. Plant nuclear safety aspects and performances assessment

The acceptability of any plant life extension program is conditioned by its safety performance objectives. In terms of requirements, the adequacy of nuclear safety has to be demonstrated by means of deterministic and probabilistic safety analyses, supplemented with ageing degradation evaluations. It is recognized that TLAA together with deterministic analyses are consistent as support documentation for PSA studies used intensively and extensively in ageing degradation management activities at any decision level. From this point of view, the assessment of plant safety performances is determined by the selected SSC behaviour during extended life period of operation. The assessment of nuclear safety characteristics involves at a great extent the compliance of defense-in-depth principle maintaining the integrity of physical barriers provided in plant design, and the accomplishment of esential safety functions.

For the defense-in-depth principle, there were evaluated the following: nuclear fuel matrix; fuel sheath and fuel channel; primary circuit pressure boundary; containment system; exclusion area.

The achievement of main safety functions were assessed considering:reactivity control; nuclear fuel cooling; radioactive material retention.

A special attention is given to the problems associated with the management of radioactive and non-radioactive wastes, of toxic and dangerous materials which are produced during plant operation, including when the life extension project activities are ongoing. During normal operation low, medium and high activity radwastes are generated, while during the extension activites only low and medium activity radwastes are produced. However, the radwastes resulting from such activities shall imply an increase of existing capacity provided for its management. This aspect includes one of the most relevant and important nuclear safety issue which the installation has to face the entire period these activities are start rolling. The approval of Regulation Board shall be sustained by a specific Environment Report relatad also with the wastes processing and treatement facilities, according to the in force rules and standards.

2.5. Safety analysis and assessment requirements during plant life extension

Safety analysis and assessment requirements during plant life extension

resulted as a consequence of the life extension activities impact on nuclear safety, licensing basis documentation and operating personnel.

Redefining the Licensing Bases Documentation is one of the major impact of life extension activities. Licensing Bases Documentation includes design, licensing and support basic documentations, according to the norms, standards, codes and design safety criteria. Usually, a plant life extension program will involve, with different weight, the updating of the LBD components.

Another impact of the plant life extension program on safety, is related to the operating personnel and offsite workers at the provided activities, in terms of potential harmfull effects. A specific report regarding the activities impact on the personnel will be issued, considering as a minimum the following: the plant refurbishment impact on operating personnel analysis, during repairs activities; the radiological impact on external personnel assessement during refurbishment and repairs activities; the radwastes amount evaluation as a result of refurbishment activities; the toxic and dangerous wastes and materials inventory evaluation resulted from refurbishment activities; the radioactive materials displaced from facility during refurbishment activities.

2.6. Environmental aspects related to plant life extension

Environmental aspects related to plant life extension concludes the importance of these activities on environment factors. The environmental impact assessment represents the process by which the direct and indirect, cumulative, main and secondary effects of a project against the environment and people's health are identified, described and established. The environmental impact assessment is an important part of the licensing process for all public or industrial projects, including the nuclear facilities.

There are detailed the refurbishments activities effects on environment, at a components level or the environment traditionally considered factors such as: land use, air quality, surface and underground waters quality, aquatic environment, terrestrial environment, as well as the life extension activities impact on socio-economic conditions. The plant operation and postulated accident conditions impact on the environment were considered too, in view of the operating experience and in force regulations requirements, including the severe accident management, intending to preserve or to improve the plant nuclear safety performances. The contributions of Polytechnic University of Bucharest – Center for Research in Energetics and Environment Protection, includes technical informations and data in a format of Environmental Report, according to US-NRC rules.

3. Conclusions

The study format and content are developed taking into account the PLiM activities complying the US-NRC rules and practices. The applicability of US-NRC PLiM rules to a CANDU project is considered beneficial from many points of view. As such:

- all the procedures related to SSC scoping and screening that require AMR and which pose a challenge to nuclear safety are identified;
- explicating the nuclear facility configuration according to LRA approach, as a sine-qua-non condition for licensing process, the elaboration of integrated plant assessment for plant design life was possible;
- a potential activities schedule, as well as the extension project main actions description were detailed;
- TLAA assessment was included in this study, and accordingly the pressurizer, reactor inlet header, pressure tube and calandria vessel were analysed;
- available industrial experience information on ageing degradation is outlined in order to qualitatively compare the methodologies and critical phenomena associated to power plant components degradation assessment, including the methods and probabilistic technics used for plant SSC condition evaluation.

The impact of SSC degradation on plant safety is evaluated considering the level of essential safety functions accomplishment, as well as the details of conformance with nuclear safety principles. The result of these analyses reveals a plant technical and safety capacity still adequate, although the degradation phenomena for an important number of SSC are directly identifiable (visual inspection) or indirectly analyzed (parametric measurement, changes of properties). As such, after 30 years of operation the nuclear facility does not experiences a full degraded status defined by inoperative and unsafe conditions. The plant process performances may be slightly reduced, but the special and support safety systems including the protective and preventive process systems may prove a minimum allowable performance level. The plant will still maintain its operability status, complying with derived requirements of nuclear safety principles and safety essential functions.

From a practical point of view, especially when a 30 years life extension is selected, the most of plant critical SSC may be repaired or changed. Therefore, it's possible that the majority of the plant modifications, for the components with qualified life, beeing already done based on rigorous program sustained by complex and complete analyses, before licensing renewal activities initiation.

REFERENCES

[1]. *** 10 CFR Part 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, 60 FR 22461, 61 FR 65175, 64 FR 72002.
[2]. *** NEI 95-10, Revision 5, Industry Guideline for Implementing The Requirements of 10 CFR Part 54 – The License Renewal Rule, Nuclear Energy Institute, January 2005.
[3]. *** NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. NRC, April 2001.
[4]. *** NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, July 2001.
[5]. *** IAEA (Draft), HWR PLiM/PLEx/ Refurbishment Process and Technology, June 2004.
[6]. *** TECDOC-1503, Nuclear Power Plant Life Management Processes: Guidelines and Practices for Heavy Water Reactors, IAEA, 2006.
[7]. *** TECDOC-1037, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Pressure Tubes, IAEA, 1998.
[8]. *** TECDOC-981, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators, IAEA, 1997.
[9]. *** TECDOC-1361, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Primary Piping in PWRs, IAEA, 2003.
[10]. *** TECDOC-1025, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings, IAEA, 1998.
[11]. *** TECDOC-1197, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Reactor Assemblies, IAEA, 2001.
[12]. *** Generic CANDU-6 type NPP Project ***, Safety Analysis Reports, 1991-2002.