http://dx.doi.org/10.5516/NET.03.2013.710

SAFETY ANALYSIS METHODOLOGY FOR AGED CANDU® 6 NUCLEAR REACTORS

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Received August 18, 2013

This paper deals with the Safety Analysis for CANDU® 6 nuclear reactors as affected by main Heat Transport System (HTS) aging. Operational and aging related changes of the HTS throughout its lifetime may lead to restrictions in certain safety system settings and hence some restriction in performance under certain conditions. A step in confirming safe reactor operation is the tracking of relevant data and their corresponding interpretation by the use of appropriate thermal-hydraulic analytic models. Safety analyses ranging from the assessment of safety limits associated with the prevention of intermittent fuel sheath dryout for a slow Loss of Regulation (LOR) analysis and fission gas release after a fuel failure are summarized. Specifically for fission gas release, the thermal-hydraulic analysis for a fresh core and an 11 Effective Full Power Years (EFPY) aged core was summarized, leading to the most severe stagnation break sizes for the inlet feeder break and the channel failure time. Associated coolant conditions provide the input data for fuel analyses. Based on the thermal-hydraulic data, the fission product inventory under normal operating conditions may be calculated for both fresh and aged cores, and the fission gas release may be evaluated during the transient. This analysis plays a major role in determining possible radiation doses to the public after postulated accidents have occurred.

KEYWORDS : Safety, Analysis, CANDU, Aging, Accidents, Predictions, Methodology

1. INTRODUCTION

All industrial plants undergo changes over time, and nuclear plants are no exception. The CANDU 6 reactor follows earlier CANDU designs such as NPD (Nuclear Power Demonstration), Douglas Point, Pickering A, and Bruce A. As such, certain aspects of plant aging were addressed directly in the design process. This was reflected in the selection of the materials, design allowances and provisions, and operating margins. In addition, certain maintenance practices have been developed over time at the stations to address the aging issues. In terms of Special Safety System availability, Reliability Studies were performed to determine the required tests and frequency of testing to ensure that necessary targets were achieved. Nonetheless, it was not possible to precisely predict how the plants would change over time at the design stage. To ensure that such changes would not compromise public safety, programs were developed to provide a continued assurance of Nuclear Station Safety.

These programs are based on the principles of monitoring/detecting, anticipating, understanding and then correcting or compensating. It has components involving Research and Development (R&D), analysis and assessment, maintenance, operational changes, and design modifications.

This paper deals with the investigation of the aging behavior of the heat transport system (HTS) for an older CANDU 6 plant as well as newer, improved plants and its potential impact on safety systems including the performance of channel flows and Critical Channel Power (CCP), one of the major components in the Regional Overpower Protection (ROP) system. This safety system protects against slow Loss of Regulation (LOR) accidents.

This paper further explores, in greater detail, post-CCP aging safety analysis methodology for a fresh core and an 11 EFPY aged core for the case of a postulated feeder stagnation break, and specifically includes fuel failure and the fission gas release during an accident transient which plays a major role in determining possible radiation doses to the public after postulated accidents have occurred.

2. CANDU 6 MAIN HEAT TRANSPORT SYSTEM AGING

Fig. 1 gives a simplified presentation of the HTS components and coolant flow of a typical CANDU 6 reactor. The HTS consists of two "figure-of-8 loops" with four main HTS pumps (P1, P2, P3, P4), four Steam Generators (SG) (B1, B2, B3, B4), and associated headers (HD) servicing the 380 reactor-core channels (ranging from channels A09 to W14). Each HTS loop consists of two HTS passes. The outlet header, purification and pressurizer interfaces are also shown.

Operational preferences as well as aging processes may cause changes to the primary HTS. These changes affect both the coolant-flow and heat transfer properties of the HTS as a whole. There are several contributing effects, some acting to increase and some to decrease the safety margins. The magnitudes of these effects vary over time, and thus the overall impact on the HTS is a complex integrated function of all mechanisms.

Operational changes can take place in a relatively short time frame, such as changes caused by utility operating preferences as well as changes in the reference analytical model interpretation caused by measurement-instrumentation calibration.

Aging related changes typically take place over relatively long time periods. The following is a list of the main known aging processes that may occur within the HTS that can influence the CCP:

- Increase in Pressure Tube (PT) diameter owing to irradiation creep (PT diametral creep): This reduces the hydraulic resistance in the channel, and hence increases its coolant-flow, but causes a redistribution of coolant flow within the bundle that can result in a reduction in dryout power. Because there is more creep in the higher power channels, there is a flow redistribution effect whereby some of the flow from the outer low power channels is redirected to the inner channels. Increased flow in the central channels mitigates the effect of PT diametral creep on the CCP for the central most important, high-power channels.
- Increase in hydraulic resistance owing to a redistribution of iron oxides (magnetite) in the HTS: Dissolution of iron and Flow Accelerated Corrosion (FAC) has been shown to occur in CANDU 6 plants. Iron is removed from the outlet feeders and is re-deposited in the cooler parts of the circuit, including the cold leg of the steam generators, the inlet feeders, and possibly the first section of the fuel channels. The magnetite layers cause both a fouling of the inside of the steam generator tubes, leading to reduced heat transfer, and also an increase in hydraulic resistance in the steam generator tubes and inlet feeders. This affects the core flow (possibly also the core top-to-bottom flow tilt), inlet header temperature, and consequently, the CCP.
- Erosion of the edges of flow-reducing orifices: This leads to relative flow redistribution from the inner to the outer parts of the reactor core.

Considerable advances have been made to mitigate these aging characteristics, by making both design and operational changes.

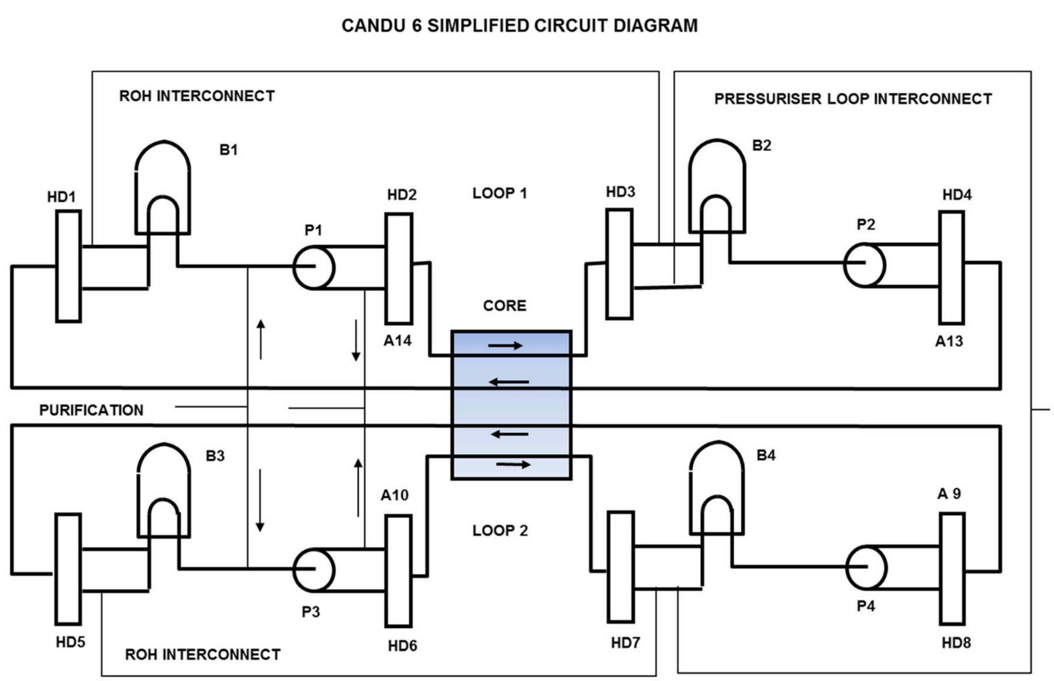

Fig. 1. CANDU 6 Simplified HTS Flow Diagram