Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 12-13, 2016

Introduction to New Regulatory Document on the Deterministic Safety Analysis in Canada (REGDOC-2.4.1)

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1. Introduction

New regulatory document on the deterministic safety analysis in Canada was published in May of 2014 with a title of "Deterministic Safety Analysis (REGDOC-2.4.1[1])" by CNSC (Canadian Nuclear Safety Commission). REGDOC-2.4.1 can be told a upgrading version of preceding Canadian regulatory documents of S-310 (2005), RD-310 (2008)[2] and GD-310 (2011)[3], which were published to replace C-6 Rev.0 (1980)[4] and C-6 Rev.1 (1999)[5] which were used as standard and basis documents for the safety analysis of existing CANDU-6 reactors including Wolsong NPP unit 1 and units 2,3,4 in Korea.

Requirements of REGDOC- 2.4.1 have great changes from those of C-6 Rev.0 and Rev.1 in terms of scope of safety analysis, acceptance criteria, analysis methods and assumption. Those changes seem to have been reflected by significance of Fukushima Daiichi nuclear accident and by international nuclear safety standards used for safety analysis of other types of nuclear reactor including PWR.

This paper will introduce some important requirements of REGDOC-2.4.1 with an emphasis on differences compared with C-6 Rev.0 and Rev.1.

2. Scope and Objectives of Deterministic Safety Analysis

As a first important difference from C-6 documents, demonstration of adequacy of provisions for protection against severe accidents and prediction of source term and doses during severe accidents were included in the objectives of deterministic safety analysis along with other objectives relevant to DBAs (Design Basis Accidents) such as to demonstrate for plant design to withstand and respond to postulated initiating events (PIEs) and to derive and confirm operational limits and conditions.

Secondly, five levels of Defense in Depth (DiD) of the plant design should be demonstrated by using deterministic safety analysis.

- Level 1: To prevent deviations from normal operation, and to prevent failures of SSC (Structure, System and Component) .
- Level 2: To detect and intercept deviations from normal operation to prevent AOOs from escalating to accident conditions, and to return the plant to a state of normal operation.
- Level 3: To minimize the consequences of accidents by providing inherent safety features, fail-safe design, additional equipment, and mitigating procedures.
- Level 4: To ensure that radioactive releases caused by severe accidents are kept as low as practicable.
- Level 5: To mitigate the radiological consequences of potential releases of radioactive materials that may result from accident conditions.

To support Level 2 and Level 3 DiD, safety analysis for AOOs (Anticipated Operating Occurrences) and DBAs are required and to support Level 4 and Level 5 DiD BDBAs (Beyond Design Basis Accidents) are analyzed.

3. Requirements for Deterministic Safety Analysis

3.1 Events to be analyzed

To identify the event to be analyzed, various NPP operating modes such as reactor startup mode, shutdown mode and normal power operation mode including low power should be considered. And events identified for the safety analysis shall include 1) component and system failures or malfunctions, 2) operator errors and 3) common-cause internally and externally initiated events.

3.2 Classification of events

The identified events shall be classified, based on the predicted frequency of occurrence from PSA and engineering judgment, into the following three classes of events, which is different from the ones of C-6 documents which has five classes of events by the predicted frequency.

- AOO: 10^{-2}/RY ≤ frequency
- DBA: 10^{-5}/RY ≤ frequency < 10^{-2}/RY
- BDBA: frequency < 10^{-5}/RY (the subset of BDBAs is referred to as Design-Extension-Conditions (DECs)

In the REGDOC-2.4.1, plant states are defined as a Fig. 1, which is very similar to one of IAEA standard documents (IAEA SSR-2/1[6]).

<table>
<tr><td colspan="2">Operational states</td><td colspan="3">Accident conditions</td></tr>
<tr><td rowspan="3">Normal operation</td><td rowspan="3">Anticipated operational occurrence</td><td rowspan="3">Design-basis accident</td><td colspan="2">Beyond-design-basis accidents</td></tr>
<tr><td>Design-extension conditions</td><td>Practically eliminated conditions</td></tr>
<tr><td>No severe fuel degradation</td><td>Severe accidents</td></tr>
<tr><td colspan="3">Design basis</td><td>Design extension</td><td>Not considered as design extension</td></tr>
<tr><td colspan="5">Reducing frequency of occurrence →</td></tr>
</table>

Fig. 1 Plant states in REGDOC-2.4.1

3.3 Acceptance Criteria

Analysis for AOOs and DBAs shall demonstrate that 1) radiological doses to the public do not exceed the established limits and 2) the derived acceptance criteria (qualitative & quantitative) are met.

The committed whole-body dose for 30 days after an event shall not exceed 0.5 mSv for AOOs and 20 mSv for DBAs. These dose limits are also changed from the dose limits of C-6 documents which have 5 distinct dose ranges for each of 5 classes of events and became stricter than the cases of C-6. However, these dose limits of REGDOC-2.4.1 would apply to new NPPs and for existing reactors the dose limits specified in the operating licenses must be met.

REGDOC-2.4.1 also provides detailed qualitative acceptance criteria for the integrity of various barriers to fission product releases and for the fundamental safety functions.

Acceptance criteria for BDBA is to demonstrate that 1) NPP meets the requirements for release limits established as the safety goals and 2) the procedures and equipment put in place to handle the accident management needs are effective, taking into account the availability of cooling water, material and power supplies.

3.4 Analysis Methods & Assumptions

According to REGDOC-2.4.1, there are three analysis method used in the deterministic safety analysis as following;

- Conservative analysis method used for Level 3 DiD
- Best Estimate plus Evaluation of Uncertainty (BEAU) used also for Level 3 DiD
- Best Estimate analysis method used for Level 2 & Level 4 DiD.

In C-6 documents, only conservative analysis method was allowed for the deterministic safety analysis. Therefore, more safety margin can be acquired by using BEAU analysis method for DBAs.

In the deterministic safety analysis for Level 3 DiD (both Conservative & BEAU), all key uncertainties in terms of modelling and input plant parameters should be identified and accounted for. However, it is not necessary for the safety analysis in support of Level 2 & Level 4 DiD to account for uncertainties to the same extent as for Level 3 DiD.

Analysis assumptions required for the analysis of AOO and DBA are as following;

- Apply the single-failure criterion to all safety systems and their support systems.
- Account for consequential failures that may occur as a result of the initiating event.
- Account for the possibility of the equipment being taken out of service for maintenance.
- Show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period.
- Credit operator actions only when there are:
  a. unambiguous indications of the need for such actions
  b. adequate procedures and sufficient time to perform the required actions
  c. environmental conditions that do not prohibit such actions

On the contrary, it is acceptable for the analysis of BDBAs to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident.

4. Conclusions

Canadian nuclear regulator (CNSC) has published new regulatory document on the deterministic safety analysis of CANDU reactors in 2014 by upgrading the preceding documents (S-310, RD-310, GD-310) published to replace C-6 and with reflecting the international nuclear safety standards and the lessons learned from Fukushima nuclear accident.

Canadian utilities are implementing the requirements of REGDOC-2.4.1 to upgrading of the existing safety analysis in a graduated manner and with a graded approach.

REFERENCES

[1] CNSC, REGDOC-2.4.1, Deterministic Safety Analysis, May, 2014.

[2] CNSC, RD-310, Safety Analysis for Nuclear Power Plants, Feb., 2008.

[3] CNSC, GD-310, Guidance on Safety Analysis for Nuclear Power Plants, Mar., 2012.

[4] CNSC, C-6 Rev.0, Requirements for the safety analysis Of CANDU nuclear power plants, 1980.

[5] CNSC, C-6 Rev.1, Safety Analysis of Candu Nuclear Power Plants, 1999.

[6] IAEA, SSR-2/1, Safety of Nuclear Power Plants (Design), Jan., 2012.