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Analysis on Radiological Consequences Following a LBLOCA of APR1400

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1. Introduction

A licensing application of Shin-Hanul Units 1&2 was submitted and is currently under KINS′ review. These units are with 1400 MWe capacity each and similar to Shin-Kori Units 3&4. The application includes radiological consequences following postulated accidents such as LOCA, SGTR, etc. In general, the most severe accident in nuclear power plants is regarded as LBLOCA when it comes to radiological impact. Therefore, this study analyzed radiological consequences following a LBLOCA of Shin-Hanul Units 1&2 using RADTRAD, a computational code for dose estimation by U.S.NRC. Assessing the important factors independently and comparing with prediction results presented in the final safety analysis report (FSAR) of Shin-Hanul Units 1&2 might be helpful in regulatory review for identifying validity of the FSAR. This paper provides RADTRAD prediction of the radiological consequences of Shin-Hanul Units 1&2 following a large break loss-of-coolant-accident and compares with licensee′s prediction results.

2. Modeling and input preparation

In this section, some of the methods used to model the Shin-Hanul Units 1&2 are described. The Shin-Hanul Units 1&2 compartment model includes source, containment, IRWST, environment, control room, etc.

2.1 APR1400 design characteristics

The containment of Shin-Hanul Units 1&2 are large and dry/wet ones and one of them is pre-stressed concrete structure to maintain its integrity for radiation shielding under the high internal pressure. The In-Containment Refueling Water Storage Tank (IRWST) installed inside the containment works as a source of water supply for emergency core cooling system, Iodine re-evolution or containment spray system. The reactor containment design pressure and temperature are 60 psig and 290℉ respectively, and cylindrical wall diameter and net free volume are 150ft and $3.128\times10^{6}ft^{3}$ respectively. The design leakage rates are 0.1% free volume per day for the initial 24 hours.

2.2 Analysis methodology

A variety of computer programs are in use for the analysis of radionuclide transport removal and dose estimation following a design basis accident. The RADTRAD computer program developed by U.S. NRC is commonly used in Korea. The licensee, KHNP, has used the same computer program to predict radiological consequences following design basis accidents. In this study, Java-based RADTRAD computer program was used to compare radiological consequences of Shin-Hanul Units 1&2. This computer program is a licensing analysis code used to show compliance with nuclear plant siting criteria for the site boundary radiation doses at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) and to assess the occupational radiation doses in the control room (CR) and/or Emergency Offsite Facility for various LOCA and non-LOCA design basis accidents (DBAs). RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at the CR, EAB and LPZ for given DBA scenarios [1]. This study utilizes up-to-date RADTRAD version 4.5.X, which was issued in January 2015 and is new NRC developed Java version of the RADTRAD code with additional features and options. Some of these include additional features and options include the reactor coolant system (RCS) activity calculator, a larger radionuclide database from ICRP-38 (838 available nuclides), and the ability for the user to model alternative source term (AST) non-LOCA DBAs described in Regulatory Guide 1.183 (RG 1.183). The accident scenario for the analysis is the limiting loss-of-coolant-accident, double-ended discharge line slot break with maximum safety injection, presented in the corresponding chapter of the final safety analysis report. Input data for RADTRAD were used from the final safety analysis report, and other variables which are not clearly stated in the final safety analysis report were decided based on the engineering judgments which are 2.5ft shielding for containment shine, 1.0ft shielding for external cloud, etc. Radiation source term is based on Regulatory Guide 1.195 and x/Q tables are from FSAR chapter 2.3. The plant power level is set at 102% and radioactive decay option is set as decay with daughter production. Gap release begins at 0 sec and the duration of accident is 720 hours. Dose conversion factors are from Federal Guidance Reports 11 and 12. Compartments in this study include containment with filter, spray and natural deposition, IRWST with Iodine re-evolution, control room with intake, inleakage, exhaust and recirculation, and environment with EAB and LPZ. The radionuclide inventory includes core inventory, RCS inventory and IRWST inventory which are described in FSAR Chapter 15. Iodine forms are aerosol 5%, elemental 91 % and organic 4% according to

RG 1.195. The containment leakage source with Iodine re-evolution is divided into sprayed and unsprayed region with mixing and Iodine removal and deposition coefficient and decontamination factor (DF) are given in FSAR Chapter 6.5.2. Iodine re-evolution is considered with spray removal in IRWST. External shine for control room dose includes containment shine from radioactive material airborne in containment and external cloud from radioactive material leaking from containment to atmosphere around control room. ESF leakage source is contributed by Iodine volatile release to ECCS equipment room and auxiliary building exhaust filter efficiency is used. The low volume purge source is from RCS liquid and vapor activity and unfiltered air leakage to environment and control room is modeled.

2.3 Prediction results

The most actual design values and parameters were used in the prediction of the radiological response to the large break loss of coolant accident. The analysis results are given in Figures 1 and 2.

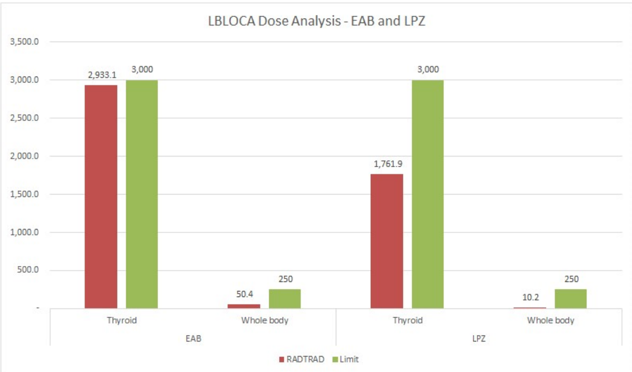
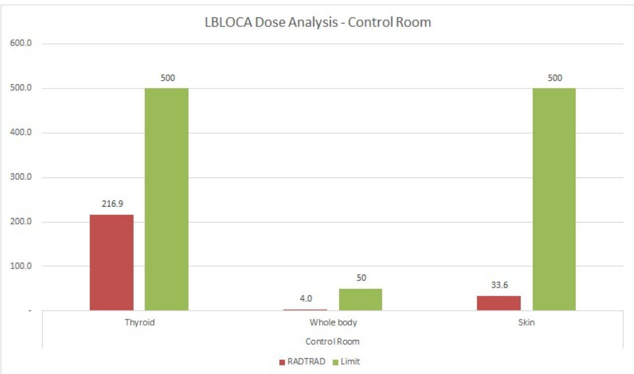


Figure 1 Dose analysis at EAB and LPZ



The dose analysis results show that thyroid doses are below the limit of 3,000 mSv and whole body doses are well below the limit of 250 mSv. Similar trend is given in control room doses where skin dose result is added. The difference of the predictions with the final safety analysis report values of dose analysis are less than 8.3%. Therefore, the predictions were almost identical to the final safety analysis report values.

3. Conclusions

Figure 2 Dose analysis in Control Room

Simulation calculations using RADTRAD can be useful for predicting radiological consequences after a loss of coolant accident in APR1400 reactors. The predictions are in good agreements with the final safety analysis report. In this study, the RADTRAD was used to calculate radiological consequences in the event of LOCA, and in the follow-up study, some non-LOCA will be analyzed with RADTRAD for the radiological consequences in APR1400 reactors. The results shown in this paper do not represent the regulatory position of Korea Institute of Nuclear Safety.

REFERENCES

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