CANDU LICENSING IN KOREA

Status Review and Future Requirements

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ABSTRACT

The licensing status and procedures, regulatory framework, and current safety issues of CANDU type reactors, Wolsong units 2, 3 and 4 are examined. Licensing difficulties and lessons learned during the safety review of Wolsong 2, 3 and 4 and future requirements are also summarized. The review was conducted, not only to confirm the design adequacy with respect to the domestic atomic laws and regulatory requirements of the vendor country, Canada, but also to reflect into the design the lessons learned from the regulatory experiences of operating Wolsong 1 to enhance the safety as high as practically possible. Safety issues observed during the licensing review, such as containment integrity, fuel channel integrity, etc., are summarized. Several efforts have been conducted to harmonize the Canadian regulations with the Korean ones by establishing domestic regulatory positions and guidelines. For example, the utility was requested to produce the CANDU safety analysis report in a standard format based on the Korean regulation which is similar to U.S. Reg. Guide 1.70. The safety of CANDU reactors in Korea will be improved and enhanced continuously through a closer cooperation with the AECB and AECL as well as with the IAEA.

1. INTRODUCTION

In Korea we have two CANDUs in operation (Wolsong 1 and 2), two under construction (Wolsong 3 and 4). Wolsong 1 and 2 have been operating since April 1983 and June 1997, respectively, and Wolsong 3 and 4 are scheduled to start commercial operations in '98, '99, respectively.

During the early stage of the nuclear power program, plants were constructed mostly on a "turn-key" contract basis, providing only a little opportunity for domestic industries to participate in the project. Domestic participation in overall construction, management, design, equipment supply, and civil construction has continuously increased since then and, at present, nuclear power technology and some related fuel cycle technology are entering into a phase of maturity.

At present, as the standardization of nuclear power plants, the development of next generation reactors, and the establishment of industrial codes and standards applicable to Korean nuclear power plants are in progress, the regulation and licensing systems in Korea are expected to be stabilized in the near future.

Korea's long-term nuclear research and development program calls for progressive design improvements for existing NPPs as well as for those to be built in the near future. Therefore, pertinent regulatory positions will be established step by step, keeping pace with the aforementioned nuclear R&D program, in terms of safety licensing and regulatory system procedures, design requirements, codes and standards, etc.

2. SAFETY REVIEW OF WOLSONG 2, 3 AND 4

Operating licenses (OLs) for Wolsong 2 and 3 were issued in November 1996 and December 1997, respectively and the OL for Wolsong 4 is scheduled for November 1998. The safety review was conducted not only to confirm the design adequacy with respect to the domestic atomic laws and regulatory requirements of the vendor country, Canada, but also to reflect into the design the lessons learned from the regulatory experiences of operating Wolsong 1 to enhance the safety as high as practically possible.

Particularly, KEPCO, the Korean utility, submitted a “Combined FSAR of Wolsong units 2, 3 and 4” as one of the licensing documents on the condition that Wolsong 3 and 4 will be designed, constructed and operated identically to Wolsong 2. Therefore, emphasis was given to confirmation of the design identity between Wolsong 2 and Wolsong 3 and 4.

2.1 Review Activities

Regulatory efforts during the construction permit (CP) stage have been focused on the following aspects; First, intensive review of the design changes/improvements compared with Wolsong 1. Second, applicability of the PWR safety issues to the units, within a practical manner, for safety enhancement equal to or above the international level. Third, design suitability and environmental effect arising from the construction and operation of multiple, (4) units at the Wolsong site.

Several domestic and foreign experts were invited to participate in the special review areas where high technology and accumulated experience were required. In addition, ACRS meetings were held occasionally to collect and reflect the experts’ opinions from research institutes and/or universities.

Particularly, an IAEA Design Review Mission was carried out by six external experts and an IAEA coordinator during the period of 22 April to 4 May, 1992. The mission did not identify any major safety issues in the design areas reviewed which would preclude the issuance of a CP for Wolsong 2. However, the mission recommended a few findings that helped the KINS to complete its safety evaluation as follows:

- Qualification of special safety system software requiring specialized expertise and availability of detailed information,
- PSA analysis extended to calculate frequencies of core damage and large off-site releases and the establishment of probabilistic safety criteria in terms of acceptable frequencies of core damage and large off-site releases, and individual risk,
- Audit calculations in selected areas of safety including PRA, reactor physics, thermalhydraulics, fuel behavior, accident analysis, containment response, and stress analysis, etc.

The final safety analysis report, operating technical specifications, radiological emergency plan, and operating quality assurance program were reviewed at the OL stage to examine the safety of the plant under normal and abnormal conditions and hypothetical accident conditions. The operating license review confirmed the final design safety of the plant and examined the operating limits and the surveillance requirements to be observed during operations. In addition, the adequacy of an emergency plan was evaluated and the safety of the public secured by providing countermeasures to mitigate undesirable situations.

Besides the safety review activities, the pre-operational inspection was performed to verify whether the nuclear facility, which has been given the CP, is constructed properly in accordance with the CP conditions and also to verify that the constructed facility demonstrates its function according to the design requirements. After the plant construction work and functional tests met the relevant requirements at each phase, the initial fuel loading and power operation commenced.

2.2 Regulatory Framework

In Korea, atomic energy laws have several levels according to their originator and applicability: the Atomic Energy Act, Enforcement Decree, Enforcement Regulation, Notice of the Minister of the MOST, and Technical Specifications which is a part of the safety analysis reports and is authorized by regulatory review. The regulatory authority for regulating the nuclear industry activities is based on the Atomic Energy Act.

As supplementary provisions, Canadian regulatory policy documents and consultative documents, and Canadian Codes and Standards are applied in the licensing review. In addition, U.S. regulations such as 10CFR and regulatory guides, SRP and NUREG are also referenced.

2.3 Safety Issues

2.3.1 Fuel Channel Integrity

To improve operational safety of the fuel channel for Wolsong units the recommended PHTS operating guidelines were developed, which provides a safe operation window in terms of pressure and temperature to maintain fuel channel integrity, which is similar to pressure-temperature curves used for primary pressure boundary integrity in PWRs and deterministic Leak Before Break (LBB) analyses were conducted using the latest understanding of the delayed hydride cracking mechanism. In addition, a comparative assessment of the LBB analysis was made using the Fitness for Service Guideline (FFSG) rev. 0 and rev. 1. An extensive AGS performance testing has been carried out during the Wolsong 2 preoperational test to measure the time needed to detect, confirm and locate leaks using simulated moisture injection.

2.3.2 Hydrogen Behaviour in the Containment

The AECB Regulatory Document R-7 and the Generic Action Item (GAI) 88-G-02 require provisions for controlling the concentration of hydrogen and/or oxygen in the containment and provide specific requirements such as assessment of the possibility of non-uniform H_2 concentration in the containment and the rationale for the number and locations of igniters. Through the review of various hydrogen sources, acceptance criteria for the number and locations of igniters, safety equipment survivability, and hydrogen concentration monitoring, etc., 44 hydrogen igniters of the hot surface type were determined to be installed in the containment.

2.3.3 Calandria Tube—Liquid Injection Nozzle Contact

Calandria tubes (CTs) are arranged in an 11.25” matrix, with liquid injection nozzles (LINs) positioned between the rows of calandria tubes.   The CT shall not be in contact with the LIN during operation. The CT/LIN contact is estimated to occur in 20-year operation by KINS’ estimation using a conservative assumption, and in 25-year operation by AECL’s. Considering 30 years for the design lifetime, CT/LIN contact is expected to occur within the design lifetime, whatever the assumptions are.

Bruce A, being the longest serving CANDU station having LINs and CTs most susceptible to sag, is the “lead station” for gap measurements. In 1993 gap measurements were conducted on Bruce unit 4 using a UT probe inserted through an existing flange on the LIN. However, CANDU 6 plants do not have such a flange so similar access would require freezing and cutting of the nozzle.

The Darlington station has funded a study into alternative methods of inspection and access via a port on the Reactivity Mechanism Deck. For CANDU 6 a similar scheme via the two existing viewing ports provides an excellent operational access opportunity for Wolsong units.

Gap measurements can be made during any convenient reactor maintenance shutdown period. However, installation data are already available as a bench mark so there is really no technical need to perform any measurements until at least 25 years after reactor start-up.

2.3.4 CANDU PSA

Level 1 PSA for both internal and external events, a level 2 PSA, and establishment of a framework for the AMP have been performed.

The common cause failures (CCFs) are determined to be important contributors to the system unavailability for both safety and non-safety systems in PWR type plants. So the level 1 PSA contains the analysis of CCFs. In addition to the consideration of CCFs, the human reliability analysis is done more systematically in the updated level 1 PSA. The scope of external events analysis is to estimate the core damage frequencies due to earthquake, internal flooding, internal fire, and other external events. The level 2 PSA covers the containment performance analysis and source term analysis. The containment performance analysis estimates the failure probability of the reactor building for a given core damage sequence. The plant-specific fragility curve for the reactor building is developed during the analysis. Wolsong-specific source terms are evaluated for the representative accident sequences.

2.3.5 Exclusion Area Boundary (EAB)

The EAB has been adopted from the beginning of the nuclear power program by applying the vendor countries' laws and regulations, i.e., U.S. and Canada. During this period, domestic laws and regulations applicable to the licensing of NPPs had not been fully developed. However each practice came from different bases and criteria.

At present, for all PWR plants licensed to date, the EAB is extended from the reactor core to a radius of 700 m. For Wolsong unit 1, the only operating CANDU reactor in Korea, the EAB is established as 914m (1000 yards) following the Canadian practice.

There are some differences between the Canadian practice and the provisions for Korean legal enforcement, especially on source term evaluation, dose assessment methodology and dose criteria. While the Notice of the Minister of MOST specified the application of the U.S. regulations, i. e. 10 CFR 100.11 and Regulatory Guide 1.4 or 1.145 in the case of determining EAB distances, the Canadian practice was applied fixed EAB distance without quantitative EAB distance evaluation based on the AECB-1079.

Korea is faced with the situation in which it is getting more and more difficult to provide appropriate sites due to small land area, high land price, high population density and public acceptance. An effort has been put into the development of a reasonable means to cope with these pressing issues including the EAB. The KINS launched a research project to develop reasonable regulatory guidelines applicable to the Korean situation since 1994 and developed a draft guideline of siting criteria for both PWRs and CANDUs in 1996. A recently qualitatively evaluated EAB of 560m was approved for the YGN 5 and 6.

2.3.6 Safety Analysis Requirement

2.3.6.1 Event Classification in C-6

The AECB's Consultative Document, C-6, provides guidance of event classifications for the accident analysis of CANDU power plants. It is required that all events be classified as Table 1 in C-6 and each event satisfy the acceptance criteria corresponding to its class. Most of the events satisfy their acceptance criteria. However, some events, such as failure of a single steam generator tube etc., can not meet its acceptance criteria, if the events are re-classified like Table 1 in C-6.

2.3.6.2 Trip Coverage

According to the Canadian regulatory requirements R-8 and R-10, there shall be at least two diverse parameters on each shutdown system. There are several accidents that do not meet the trip effectiveness requirements of R-8 and R-10, such as a small break LOCA, pressure tube rupture, feedwater line break etc. It was stated that exceptions to this requirement are considered to be acceptable for limited configurations; namely if providing two parameter coverage is either: (1) not practicable (that is either technically not feasible, or if the only means of detection would be a setpoint within normal operating conditions), (2) counter-productive to public safety, or (3) a crushing economic burden. Exceptions may be permitted if the single trip coverage regions cannot be removed with the new trip parameter instead of the existing ones. In this case, it should be justified that incorporation of a second parameter protection against an event is 'impracticable' after the analysis results for the new trip parameters considered shall be reviewed.

2.3.7 Validation and Verification of CATHENA and PRESCON2 Codes

CATHENA, a thermalhydraulic system code, and PRESCON2 a containment thermalhydraulics code, have been reviewed in the area of code structures, models and correlations. Independent system codes and methodology, RELAP5 and CONTAIN also have been developed in order to review the adequacy for the analysis results and to justify the quality and adequacy of the system analysis computer codes used by the licensing applicant.

Critical ROH and RIH break LOCAs, small break RIH LOCAs and Thermosyphoning tested in the RD-14 facility have been assessed with RELAP5/MOD3 and compared with CATHENA. CATHENA analysis results for ROH LOCA and Main Steam Break in Wolsong FSAR were also compared with those with RELAP5. A containment analysis with CONTAIN1.12 was performed and compared with the PRESCON2 results in the Wolsong FSAR.

A brief validation guideline of the code system has been developed and applied to the review of the Wolsong units 2-4 FSAR. Detailed and definite guidelines for the code approval will be provided through more systematic procedure in the near future.

2.3.8 Leaktight Integrity of Containment

The leaktightness of containment of CANDU-6 plants is ensured by a prestressed concrete structure and a rigid type non-metallic liner applied on the inside concrete surface. The lessons learned from the failure of ILRT requirement of CANDU-6 plants which had been appropriately designed and constructed according to related codes and specifications require the improvement of liner material and related technical standards including the strengthening of qualification test provisions and periodic inspection requirements for containment.

In addition to the integrity on LOCA pressure which can be confirmed by ILRT, when the integrity on ultimate pressure (dual failure condition) is concerned, the design change of the liner material to steel that is ductile to large deformation and high leak resistance has to be considered. Also, the strengthening of qualification test provisions is required with consideration of concrete crack bridging capability and/or aging effect.

2.3.9 Integrity of Spent Fuel Storage Bay (SFSB)

Water has been collecting in the sump of Wolsong 1, designed for collection of cooling water leaking from the SFSB, with maximum rate of 8 Mg/Day. A similar phenomenon has been reported in other CANDU-6s.

The SFSB is surrounded by a water proof membrane which prevents the release of leaked cooling water from the SFSB to the ground. The inflow of water to the sump located between the SFSB and the membrane indicates the leakage of cooling water from the SFSB or the loss of function of the membrane.

As the leakage of cooling water from the SFSB causes the direct release of radioactive material to the environment, the collaboration between CANDU-6 owners is required to resolve this matter in consideration of detailed inspection, necessity of repair and future design changes.

Besides the aforementioned safety issues, the followings issues are also reviewed:

- Fuelling machine design by seismic qualification,
- PDC hardware/software development plan and test results,
- Confirmation of containment integrity: containment isolation by crimping,
- CIV leakage testing,
- Design adequacy of main steam isolation valves,
- Feeder wall thinning.

3. LICENSING DIFFICULTIES AND LESSONS LEARNED

3.1 Differences in Regulatory Procedures and Practices

Laws and regulations between Korea and Canada are quite different. Korean laws and regulations have a prescriptive approach similar to those of the U.S. such as application of established codes and standards, technical criteria and procedures including two-step licensing, while Canadian laws and regulations are a consultative one with two-step licensing. Based on the skills and technical consultation between the regulatory body and the nuclear industry, the safety of plant design is reviewed continuously even after issuance of the license.

3.2 Standard Format of Safety Analysis Reports

CANDU Safety Analysis Reports (SARs) are written according to the format and other prescriptions of the AECB, reflecting the Canadian regulatory system, which is substantially different from that of the U.S. Reg. Guide 1.70; particularly different are the SAR format, content and description. From the viewpoint of Korea's regulatory review experience with U.S. PWRs, CANDU SARs looks more like they are written in favor of the Canadian regulator's convenience in reviewing design safety. Therefore, in Korea, the format and content of SARs are required to follow the specifics prescribed in U.S. Reg. Guide 1.70 from Wolsong unit 2.

3.3 Standard Review Plan

The Standard Review Plan (SRP) was prepared for the guidance of staff reviewers in performing safety reviews of applications to construct or operate nuclear power plants in the U.S. The principal purpose of the SRP is to assure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the staff review process to interested members of the public and the nuclear power industry. However, there is no information in the Canadian regulatory system which is similar to the U.S. SRP so that it is difficult to assure the qualified regulatory review based on well-defined procedures or it may take much time and effort to prepare the SRP for CANDU plants.

3.4 Regulatory Requirements for the CANDU System

Current regulatory requirements for the safety analysis of the CANDU system are given as C-series and R-series, such as C-6, R-7, R-8, R-9, and R-10, etc. Consultative document C-6 describes the requirements for safety analysis of CANDU plants. Some of the requirements are not clearly defined and described in detail, compared with the U.S. regulatory requirements, so it is often difficult to apply these as regulatory guidelines.

In Canada, there is no regulatory system to confirm the validation and verification of the computer codes used in the accident analysis. Moreover, there are no independent audit activities to provide assurance that the design analyses performed by the supplier of the NPPs had been accomplished in a reasonable and complete manner.

These days, we can say that it is a worldwide trend to make regulatory requirements clear and well-defined, well-based and well-documented. And also from the public acceptance point of view, the general public in Korea asks for a regulation according to the prescriptive requirements.

Therefore, it would be an important task to establish a regulatory framework which can be applied to the regulation of CANDU plants uniformly across the world through the development of a standard format for Safety Analysis Reports (SARs), Standard Review Plans (SRPs), regulatory guides and application criteria of appropriate industrial codes and standards.

4. FUTURE REQUIREMENTS

4.1 Resolution and Implementation of Generic Licensing Issues

Generic licensing issues are the safety issues associated with Canadian CANDU nuclear power plants that applied to all or several plants and were considered insufficiently resolved. Therefore, each of the identified generic issues should be resolved before a new nuclear power plant license is issued, but none are considered sufficiently critical to require refits on existing plants.

4.2 Implementation of TMI Action Items

Some Canadian responses and provisions to the U.S. developed TMI action plan/items were reflected in the review. The AECB issued a TMI implementation plan for CANDU reactors as INFO-0003 (TMI-A Review of the Accident and Its Application for CANDU Safety), and some items have been implemented at Wolsong 1, 2, 3 and 4. Therefore, a systematic implementation guideline of TMI action items needs to be provided for future CANDU reactors.

4.3 Severe Accident Prevention and Mitigation

A high degree of redundancy in heat sink capability should be incorporated in future CANDU reactors to ensure both a low severe core damage frequency and to ensure two redundant means of heat removal for any external events such as earthquakes, etc. Eventually, for a severe accident, the related systems would minimize the challenge to the containment system.

4.4 Design Improvement Based on Operating Experiences

The operating experiences gained at Wolsong 1 and CANDU reactors in Canada should be investigated systematically and the resulting recommendations should be implemented.

4.5 Application of the Standard Review Plan (SRP)

The SRP for CANDU reactors is currently being developed at KINS and the utility may be required to follow the detailed, specific requirements in the SRP.

4.6 Advanced Design to Improve Safety

A progressive improvement in the safety of nuclear power plants has been set as a national policy in Korea. Therefore, it may be recommended to establish utilities' design requirements for future CANDUs such as EPRI's "Utility Requirement Documents" for advanced and evolutionary PWRs. The improved design should be proved by existing well-established safety concepts and by intensive research and development. The safety of the new or improved designs should be confirmed not only in the design itself but also in the overall effects on the inter-related systems.

5. CONCLUSIONS

Korea has carried out a very ambitious nuclear power program since the early 1970s with a strong commitment to nuclear power development as an integral part of the national energy policy. However, the diversification of reactor types and vendors, such as the U.S., Canada, and France has caused some difficulties in regulation and licensing of those nuclear power plants in Korea. Particularly, for Wolsong 2, 3 and 4, several efforts have been conducted to harmonize the Canadian regulations with the Korean regulations by establishing domestic regulatory positions and guidelines. Continuing efforts on the development of SRPs and computer codes, etc. will be incorporated to accomplish a more systematic regulatory process through close cooperation with the AECB and AECL as well as with the IAEA. Finally, the regulatory requirements for future CANDU reactors are suggested with the aim to ensuring that the nuclear power plants will be safer and more reliable.