CANFLEX-NU FUEL LICENSING STATUS AND ISSUES IN KOREA

GEUN-SUN AUH, JUN-SANG PARK, SWENG-WOONG WOO, YONG-HO RYU

Korea Institute of Nuclear Safety

ABSTRACT

The CANFLEX-NU Fuel Design Report (FDR) for Wolsung 1,2,3,4 was submitted for licensing review in July 1996. The FDR contains sections of fuel rod design, fuel bundle design, nuclear design and thermal-hydraulic design. Each section describes the design bases, design methodology and design evaluation results showing that the design bases are met. The CANFLEX-NU fuel design is not finalized yet in Korean licensing point of view. For example, among others, new Xc-BL correlation is needed to be developed, fuel rod gap reduction effect is to be considered in the Critical Heat Flux, more information for power ramp defect of fuel rod especially in the end-cap weld region is needed in the fuel rod design, and enough data are not available in irradiated conditions in the fuel rod and bundle designs. The specific detailed technical licensing issues and their backgrounds are explained for the CANFLEX-NU FDR in Korea. The Korean nuclear regulation environment is changing due to the Korean government policy of overall regulation reduction. The contents and timetables of Korean nuclear regulation changes are also explained for new fuel licensing.

1. INTRODUCTION

Per Article 44.3 of the Atomic Energy Act, the CANFLEX-NU Fuel Design Report (FDR) for Wolsung 1, 2, 3, 4 was submitted to the Ministry of Science and Technology (MOST) for licensing review and approval in July 1996. The KINS (Korea Institute of Nuclear Safety), which is entrusted with regulatory works by law and the government, and performs detailed assessment of the technical submissions and inspection of nuclear facilities as a technical expert group, received the CANFLEX (Canadian Flexible Fuel Bundle)-NU (Natural Uranium) FDR for its licensing review on July 5, 1996. The CANFLEX-NU fuel bundle is different from the existing 37-element (13.1 mm element) bundle. The CANFLEX-NU fuel bundle is composed of 43 elements with two different size elements (11.5 mm outer elements and 13.5 mm inner elements). The outside diameter of the CANFLEX-NU fuel bundle is the same as that of the 37-element bundle in order to assure the compatibility with the fuel channel and the fuel handling system. In fuel performance point of view, two major design changes were made. They are CHF (Critical Heat Flux) enhancement buttons and flattened radial power profile within a fuel bundle (resulting in about 20% lower fuel rating) through two different diameter elements. The CHF enhancement design without large pressure drops plays main role of giving higher critical channel power than the 37-element bundle owing to the buttons and more fuel rods compared with the 37-element fuel bundle [1]. The flattened radial power profile within the CANFLEX-NU bundle gives lower element power rating and more homogeneous enthalpy distributions in the subchannels than the 37-element fuel bundle, and improves the fuel mechanical and thermal-hydraulic performances [1].

2. FUEL DESIGN REPORT

The FDR [2] of the CANFLEX-NU fuel contains sections of fuel rod design, fuel bundle design, nuclear design and thermal-hydraulic design. Each section describes the design bases, design methodology and design evaluation results showing that the design bases are met.

The fuel rod design includes descriptions of fuel rod internal pressure, fuel pellet temperature, cladding oxidation, stress corrosion crackling at circumferential cladding ridge, stress corrosion cracking at endplug/cladding welded zone, fuel rod fatigue, cladding collapse into axial gap, formation of longitudinal ridge, and geometric stability of fuel rod.

The fuel bundle design includes descriptions of fuel pressure drop, fretting of pressure tube and inter-element spacer, end plate fatigue, pressure tube corrosion, fuel bundle/pressure tube interaction, fuel bundle/pressure tube rolled joint interaction, shield plug/fuel bundle string clearance, compatibility with fuelling machine, axial loads applied during refueling, refueling impacts, crossflow fretting in the liner tube, and demonstration irradiation in experimental/commercial reactors.

The nuclear design includes descriptions of neutron spectrum of the CANFLEX-NU lattice, multiplication factor, void coefficient, reactivity coefficient, incremental cross section, reactivity of the control system and power distribution in the CANFLEX-NU bundle.

The thermal-hydraulic design includes descriptions of pressure drop of the fuel bundle string, critical power ratio, flow stability, channel flow rate and fuel temperature.

3. LICENSING FRAMEWORKS OF NEW FUEL DESIGN

The design, qualification and implementation of CANFLEX-NU bundles can be categorized into the following 3 phases in Canadian licensing environment [3]:

- Phase 1: Conceptual development, Out-reactor and In-reactor testing
- Phase 2: Power reactor irradiation of 24 bundles
- Phase 3: Transition from a 37-element bundle core to a CANFLEX-NU bundle core via normal fuelling

Based on the past Canadian practices, licensing submission and approval is needed for power reactor irradiation after phase 1 and phase 2. The licensing submission for irradiation of 24 CANFLEX-NU bundles may contain information on background, overview and summary, power-reactor irradiation program, Design Manual, Design Drawings, Fabrication Report, and Safety Case. The Safety Case for CANFLEX-NU demonstration irradiation includes review of all accident scenarios in the safety report, identification of accidents affected by the fuel design change, calculations to quantify differences in accident behaviors between CANFLEX-NU and 37-element bundles, and evaluation on the effects for the accidents identified. It can be said that the licensing submission before 24 bundle irradiation centers on design verifications, and safety implications that are related to the 24 bundles irradiation, and the licensing submission for the core conversion to the CANFLEX-NU bundles centers on the safety assessments of the transition core and the full CANFLEX-NU core.

Two licensing submissions and approvals are needed for power reactor irradiation of new design fuel in Korea. The Atomic Energy Act [4] provides the legal foundation for nuclear activities in Korea. The Article 44.3 of the Atomic Energy Act says, "The nuclear fuel cycle vendor shall obtain the approval of the Minister of Science and Technology on the fuel design and fabrication methods before the start of the fuel fabrication." Per the above Article of the law, the CANFLEX-NU FDR and Fabrication Method Report were submitted to the KINS via the MOST. The FDR is to obtain the MOST approval to start the fuel fabrication and to prepare for next licensing stage. For full implementation of new fuel in a power reactor, another licensing and approval is needed for the Operating License (OL) change via the FSAR (Final Safety Analysis Report) change per Article 21 of the Atomic Energy Act. It is noted that all the design, analysis and tests including in-reactor and out-of-reactor prototype testing should be completed and documented for the OL change submission with the safety analyses of the transition core and the full CANFLEX-NU core. Per recent amendment (dated February 8, 1999) of the law, the Article 44.3 will disappear on August 9, 1999 due to the Korean government policy of overall regulation reduction. In the future, it is expected that the nuclear fuel cycle vendor prepare and submit "Specific Technical Topical Report" on the new fuel design for the approval of the MOST per the Article 104.2 of the law. The Specific Technical Topical Report for new fuel design in Korea may be prepared in the future reflecting a Topical Report of new fuel design in the United States. The Topical Report contains Design Bases, Design Methods, Design Evaluation results of analyses and tests, and generic Safety Analysis results. It also includes irradiation plans of Lead Test Fuel Assemblies for power reactor irradiation experiences.

4. LICENSING REVIEW STATUS IN KOREA

The KINS received the CANFLEX-NU FDR and Fabrication Method Report on July 5, 1996. Three KINS staff visited the AECL on October 1996 to discuss with AECL staff with a guidance of the KAERI (Korea Atomic Energy Research Institute), which is the present CANFLEX-NU nuclear fuel cycle vendor in Korea. There were 4 major questionnaires totaling more than 166 questions. There were also many meetings between KINS and KAERI staff on the questions in the especially late 1998 and early 1999. Most of the questions were successfully answered.

In licensing review on the mechanical design of the CANFLEX-NU fuel, it was confirmed that most of the design features satisfied the corresponding design criteria. However, there are some unconfirmed design features: (1)The "SCC (Stress Corrosion Cracking) Defect Threshold Power History", an empirical correlation derived from the database of the 28- and the 37-element bundles, was used as the SCC fuel defect criteria at the circumferential ridge of the CANFLEX-NU fuel element. In principle, the above empirical correlation can not be applied to CANFLEX-NU fuel element because the dimension of the CANFLEX-NU fuel element is not within the ranges of database of that correlation. To demonstrate the applicability of that empirical correlation to the

CANFLEX-NU fuel element, the comparative analysis result of the INTEGRITY code showing the similarity of the SCC defect threshold conditions between the 37-elements and the CANFLEX-NU fuel was provided. Considering the minor change of fuel element dimension and the analysis result by the INTEGRITY code, it is considered that the application of the empirical correlation to the evaluation of SCC defect of the CANFLEX-NU fuel element is not inadequate. However, it is thought that enough operating experience data in commercial reactors should be provided to show the adequacy of the CANFLEX-NU fuel element design to the SCC defect during power ramp conditions. (2) Initially, a specific stress limit was used as a design criterion for the SCC defect analysis at the cladding/endcap weld region of the CANFLEX-NU fuel element, and later, it was replaced by experimental evaluation requiring the power ramp tests. However, an outline of the power ramp test result for only 1 bundle irradiated in the NRU research reactor was provided. It is considered that sufficient and detailed in-reactor test results encompassing the various power histories in commercial reactors should be provided to demonstrate the validity of the CANFLEX fuel element design for the SCC defect at cladding/endcap weld region. (3) Fuel rod bowing needs not be limited to a specific value, but it must be included in the thermal-hydraulic analysis of the fuel bundle. Quantitative evaluation and analysis of the fuel rod bowing should be based on Post-Irradiation Examination (PIE) of the fuel elements. In the FDR licensing review, however, quantitative analysis results for the CANFLEX fuel rod bowing were not provided.

For a new design fuel, demonstration irradiation and PIE must be performed to confirm the validity of the fuel design (fission gas release, irradiation growth of fuel bundle etc.) and to demonstrate satisfactory fuel performances in reactor conditions (power ramp defect, fuel rod bowing, compatibility of the irradiated fuel bundle with the fuel channel and the fuel handling system etc.). In the FDR licensing review, however, no detailed results on demonstration irradiation and PIE for the CANFLEX fuel were provided, although demonstration irradiation for 5 test bundles in the NRU research reactor had been performed. Moreover, information or plan on demonstration irradiation in commercial reactors was not provided. In summary, there are 3 licensing open items of the following in the mechanical design of the CANFLEX-NU fuel:

- Submission of the PIE results for the 5 test bundles irradiated in the NRU research reactor.
- Submission of the PIE results or the demonstration irradiation program of the CANFLEX fuel in commercial reactors

- Quantitative analysis of the CANFLEX-NU fuel rod bowing

The purpose of the fuel channel thermal-hydraulic design is to determine the heat removal capability in all channels and to meet performance and safety criteria. The NUCIRC [5], which is a steady-state thermal-hydraulic analysis computer code, is used for this to simulate the Heat Transport System of Wolsung-1. Estimation of the code includes channel flow rate, physical status of the coolant and fuel rod dryout power. The minimum CPR (Critical Power Ratio) was presented to preclude dryout at the fuel rod surface during the steady-state and slow Loss of Reactivity Control. The critical heat flux of the CANFLEX-NU was obtained from the test results of Freon-12, 22 and 134a without test results of H_2O. From the comparative results of limited Freon tests, it has been shown that the CANFLEX-NU bundle is better than the 37-element bundle in the critical heat flux. The FDR proposed to use the existing Xc-BL correlation for the CANFLEX-NU fuel. However, the limited CHF test of Freon is not enough to allow unlimited use of the existing Xc-BL correlation. Therefore, new XC-BL correlation has to be developed with enough test data of H_2O to cover wide-ranges of physical parameters including various power distributions and fuel rod gap reduction. The NUCIRC-MOD1.505 was used to design the NUCIRC-NU, but it was not validated against CANFLEX-NU test data of H_2O. Based on the above descriptions, we have following 3 technical licensing open items in the thermal-hydraulic design area along with two obvious considerations of "(1) appropriate evaluation has to be performed for the reactors which have different reactor thermal-hydraulic parameters than the Wolsung-1 original design and (2) to meet the minimum channel flow rate design requirement, Reactor Outlet Header has be operated to have less than 4% steam quality during normal reactor operation":

- new Xc-BL correlation based on H_2O test results with enough test data ranges, distributions and numbers with conservative considerations of radial flux distribution and non-uniform axial power distribution,
- Critical Heat Flux effect of fuel rod gap reduction due to bearing pad wear, manufacturing tolerance and fuel rod bow,
- Validation of NUCIRC Code using the CHF test results of H_2O to evaluate thermal-hydraulic design effects of the validation.

5. CONCLUSIONS

The CANFLEX-NU Fuel Design Report (FDR) for Wolsung 1,2,3,4 was submitted for licensing review in July 1996. The FDR contains sections of fuel rod design, fuel bundle design, nuclear design and thermal-hydraulic design. Each section describes the design bases, design methodology and design evaluation results showing that the design bases are met. The CANFLEX-NU fuel design is not finalized yet in Korean licensing point of view. For example, among others, new Xc-BL correlation is needed to be developed, fuel rod gap reduction effect is to be considered in the Critical Heat Flux, more information for power ramp defect of fuel rod especially in the end-cap weld region is needed in the fuel rod design, and enough data are not available in irradiated conditions in the fuel rod and bundle designs. Except for the above items, the CANFLEX-NU FDR contains satisfactory design results in Korean licensing point of view.

6. REFERENCES

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[2] Suk, H.C. and et. al., “CANDU-6 CANFLEX-NU Fuel Design Report,” KAERI/TR-1220/99, February 1999.

[3] Lau, J., “Safety Case to Support CANFLEX Demonstration Irradiation,” AECL-Sheridam Park, October 25, 1996.

[4] “Atomic Energy Act,” Republic of Korea, dated December 13, 1997 and February 8, 1999.

[5] “NUCIRC-MOD1.505, User’s Manual,” TTR-516, December 1993.