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A Core Design Study for a Small Modular Boiling Water Reactor with Long-Life Core

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Abstract — *This paper presents the core design and performance characteristics of the Novel Modular Reactor (NMR-50), a 50-MW(electric) small modular reactor. NMR-50 is a boiling water reactor with natural-circulation cooling and two layers of passive safety systems that enable the reactor to withstand prolonged station blackout and loss of ultimate heat sink accidents. The main goal in the core design is to achieve a long-life core (~10 years) without refueling for deployment in remote sites. Through assembly design studies with the CASMO-4 lattice code and coupled neutronics and thermal-hydraulic core analyses with the PARCS and RELAP5 codes, a preliminary NMR-50 core design has been developed to meet the 10-year cycle length with an average fuel enrichment of 4.75 wt% and a maximum enrichment of 5.0 wt%. The calculated fuel temperature coefficient and coolant void coefficient provide adequate negative reactivity feedbacks. The maximum fuel linear power density throughout the 10-year burn cycle is 18.7 kW/m, and the minimum critical power ratio is 2.07, both of which meet the selected design limits with significant margins. Preliminary safety analyses using the RELAP5 code show that the core will remain covered during the entire transient procedure of two design-basis loss-of-coolant accidents. These results indicate that the targeted 10-year cycle length is achievable while satisfying the operation and safety-related design criteria with sufficient margins.*

Keywords — *Boiling water reactor, long-life core, small modular reactor.*

Note — *Some figures may be in color only in the electronic version.*

I. INTRODUCTION

In the past few years, small modular reactors (SMRs) have drawn a great surge of research and development interest in both the United States and other countries around the world.[1–4] Compared to conventional large monolithic power reactors, SMRs, as implied by the name, have two essential characteristics: The reactors are "small" in the sense of both physical size and power output; the reactors are to be deployed modularly, that is, multiple units of the reactor can be constructed and operated sequentially and independently to gradually meet the power demand of the operating site. The low power output and the physical smallness of SMRs derive benefits in areas of safety, fabrication, operations, and economics. The modular design features of SMRs provide substantial flexibility to the deployment of the SMRs. The main parts of SMRs can be assembled in a factory and transported to remote sites, which would further promote the deployment of SMRs to isolated or undeveloped areas where mature infrastructure for electricity transmission is not available, such as developing countries, rural areas, and military bases. It is foreseen that SMRs will be prominently promoted in the near future.[1]

There are many types of SMRs proposed in the market at this time.[1] Among them, almost all the light water–cooled SMRs, such as NuScale,[5] mPower[6] and the International Reactor Innovative and Secure[7] (IRIS),

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adopt an integral pressurized water reactor (PWR) design concept. In the integral PWR design, the steam generator (SG) and the pressurizer are housed within the reactor vessel such that not only the number of key reactor components but also the possible failures associated with these components are reduced. Therefore, these designs are claimed to be simpler and safer.[5–7] However, this claim remains to be proven as the integral features impose additional technological challenges to the reactor vessel itself, which is the single most important component in a nuclear power plant. Additionally, PWR-type SMRs, either with natural or forced circulation in the primary system, commonly feature a large, tall reactor pressure vessel (RPV) due to the space required for the integrated SG and pressurizer and the need for a large thermal driving head associated with the single-phase natural-circulation flow. Large-sized containment and other components such as the reactor building inevitably accompany a large RPV. In contrast, an integral boiling water reactor (BWR)-type SMR can feature fully passive safety and coolant circulation systems by taking advantage of a two-phase flow driving head, which results in a much smaller and much simpler RPV because the coolant circulation pumps and pressurizers within the reactor vessel are eliminated. Moreover, a BWR-type SMR has no need for a SG, which further reduces the complexity of the system and eliminates the safety and operation and maintenance (O&M) concerns associated with SG tube failures. This also improves the economic viability and yields a high energy conversion efficiency as it bypasses the intermediate heat exchanger system.[8]

Taking advantage of these characteristics of a BWR, researchers at Purdue University are designing the Novel Modular Reactor[8] (NMR-50), a 50-MW(electric) BWR-type SMR, with the design objective to achieve a cycle length of ~10 years. The NMR-50 design is based on the SBWR-600 (Ref. 9) design of General Electric (GE) and the SBWR-200 (Ref. 10) design of Purdue University with renovations to encompass SMR concepts and enhanced safety systems. In order to reduce construction cost the RPV height was minimized within the limit of maintaining the natural-circulation capability in the vessel. Using the three-level scaling method developed by Ishii et al.,[8] the axial dimensions were reduced by half relative to the SBWR-200 design while retaining the radial dimensions. As a result, the core volume was reduced by half while the rated thermal power was reduced by a factor of 4 [from 660 to 165 MW(thermal)], and thus, the power density was reduced to half of SBWR-200.

As part of the NMR-50 design efforts, core design studies were performed to develop a core design to yield a 10-year cycle length with a fuel enrichment within the industrial limit of 5 wt% while satisfying other operation and safety-related design constraints. Assembly design studies were performed using the CASMO-4 lattice code,[11] including the generation of homogenized assembly cross sections and form functions for subsequent whole-core calculations. Coupled neutronics and thermal-hydraulic (T/H) core analyses were performed using the PARCS (Ref. 12) and RELAP5/MOD3.3 (Ref. 13) code systems.

In this paper, we present the core design and performance characteristics of NMR-50. In Sec. II, the overall design concepts of NMR-50 are briefly summarized. The core design objectives and constraints are presented in Sec. III, and the design approaches and analysis methods are described in Sec. IV. The resulting core design and performance characteristics are discussed in Sec. V. Section VI presents the preliminary safety analysis results for two design-basis loss-of-coolant accidents (LOCAs), and Sec. VII provides some conclusions.

II. NMR-50 DESIGN CONCEPTS

In contrast to other SMR designs, NMR-50 features many advantages including the two-phase natural circulation of coolant, a compact and simplified RPV design, a better energy conversion efficiency, a long-life core, and a reduced need for alternating-current power.[8] All these salient features would make NMR-50 one of the most economical designs in terms of both initial construction and O&M costs, which are important factors in determining the deployment of a SMR. As the NMR-50 design is based on proven simplified BWR technology, there are almost no obvious technological and regulatory obstacles preventing it from prompt development after proof of principal is demonstrated. Some key design parameters of NMR-50 are compared in Table I to the aforementioned integral PWR-type SMRs. It can be seen that the height of the RPV of NMR-50 is greatly reduced compared to other SMR designs, which consequently cuts off the manufacturing cost of NMR-50. The cycle length of NMR-50 is also significantly longer than the other reactors although the fuel enrichment is similar.

To address the lessons learned from the recent Fukushima nuclear accident, NMR-50 encompasses passive safety design concepts. In order to cope with design-basis accidents (DBAs) and beyond DBAs, NMR-50 integrates two layers of passive safety systems that could provide adequate removal of decay heat for an indefinite period without outside intervention.

The first layer of passive safety systems consists of conventional passive safety systems such as an automatic depressurization system (ADS), a passive containment

TABLE I

Comparison of Key Design Parameters of Several Light Water Reactor–Based SMRs

SMR		NMR-50	NuScale (Ref. 5)	mPower (Ref. 6)	IRIS (Ref. 7)
Type		Simplified BWR	Integral PWR	Integral PWR	Integral PWR
Primary coolant system		Two-phase natural circulation	Single-phase natural circulation	Forced circulation	Forced circulation
Rated power [MW(electric)]		50	45	180	335
Primary system pressure (MPa)		7.17	8.72	14.1	15.5
Reactor vessel	Height (m)	8.5	13.7	22	21.3
	Diameter (m)	3.48	2.7	3.6	6.78
Cycle length (years)		~10	2	4	2.5 to 4
Fuel enrichment (%)		5	<4.95	5	4.95

cooling system (PCCS), and an isolation condenser system (ICS). In the event of low water level in the RPV, the suppression pool (SP) water is available to flood the reactor core. The gravity-driven cooling system of SBWR-600, however, is eliminated, but this loss is compensated by increasing the core inventory and simplifying the design, which further enhance the robustness of the passive safety system.

The second layer of passive safety systems includes a passive containment cavity cooling system and a long-term cooling water storage pond. When the primary safety system becomes unavailable in a prolonged station blackout or a loss of ultimate heat sink accident, the secondary safety system is activated, and the cavity between the steel containment and surrounding concrete shield is flooded with water from the long-term storage pond by gravity.

The two layers of passive safety systems also follow the passive safety system requirements from International Atomic Energy Agency guidelines in terms of passivity, redundancy, and diversity. The first layer of the passive safety systems consists of most conventional BWR passive safety components while the second layer of the passive safety system supplies redundant and diverse safety components for the system. With the passive safety systems, NMR-50 can withstand a prolonged station blackout and a loss of ultimate heat sink accident. This is extremely important for the deployment of SMRs in remote areas or developing countries where a well-established infrastructure is not available. A schematic view of the passive safety systems of NMR-50 is shown in Fig. 1.

III. CORE DESIGN OBJECTIVES AND CONSTRAINTS

As part of the NMR-50 design efforts, core studies were performed to develop a core design in accordance with the overall design objectives. The primary core design goal for NMR-50 was set for a long-life core in order to facilitate its specific applications to remote sites and small grid systems, where industrial infrastructures are underdeveloped and unit cost of electricity generation is very high with conventional technologies. A long cycle length would also reduce the cost associated with refueling shutdown and spent fuel storage.

The specific core design goal of the NMR-50 core was to achieve a 10-year core lifetime with no refueling and fuel enrichment within the industrial limit of 5 wt% while satisfying other operation and safety-related design constraints. Negative fuel temperature and coolant void reactivity coefficients as well as a sufficient shutdown margin were considered as the reactivity-related design constraints. In order to limit the peak cladding temperature during the LOCA and to avoid the coolant dryout condition, two additional thermal design constraints were imposed on the maximum fuel linear power density (MFLPD) and the minimum critical power ratio (MCPR). The main core design objectives and constraints of NMR-50 are summarized in Table II. The specific design limits on MFLPD and MCPR were adopted from the reference SBWR-600 design.[9]

IV. DESIGN APPROACHES AND ANALYSIS METHODS

As mentioned in Sec. I, the NMR-50 core configuration was initiated by reducing the axial dimensions of SBWR-200 by half while retaining the radial dimensions. As a result, the average power density was reduced by half, which in turn reduces the specific power by half for the same fuel volume fraction in the fuel assembly. The reduced specific power naturally allows a longer cycle length for the same burnup. In order to increase the cycle length further, a single-batch fuel management scheme

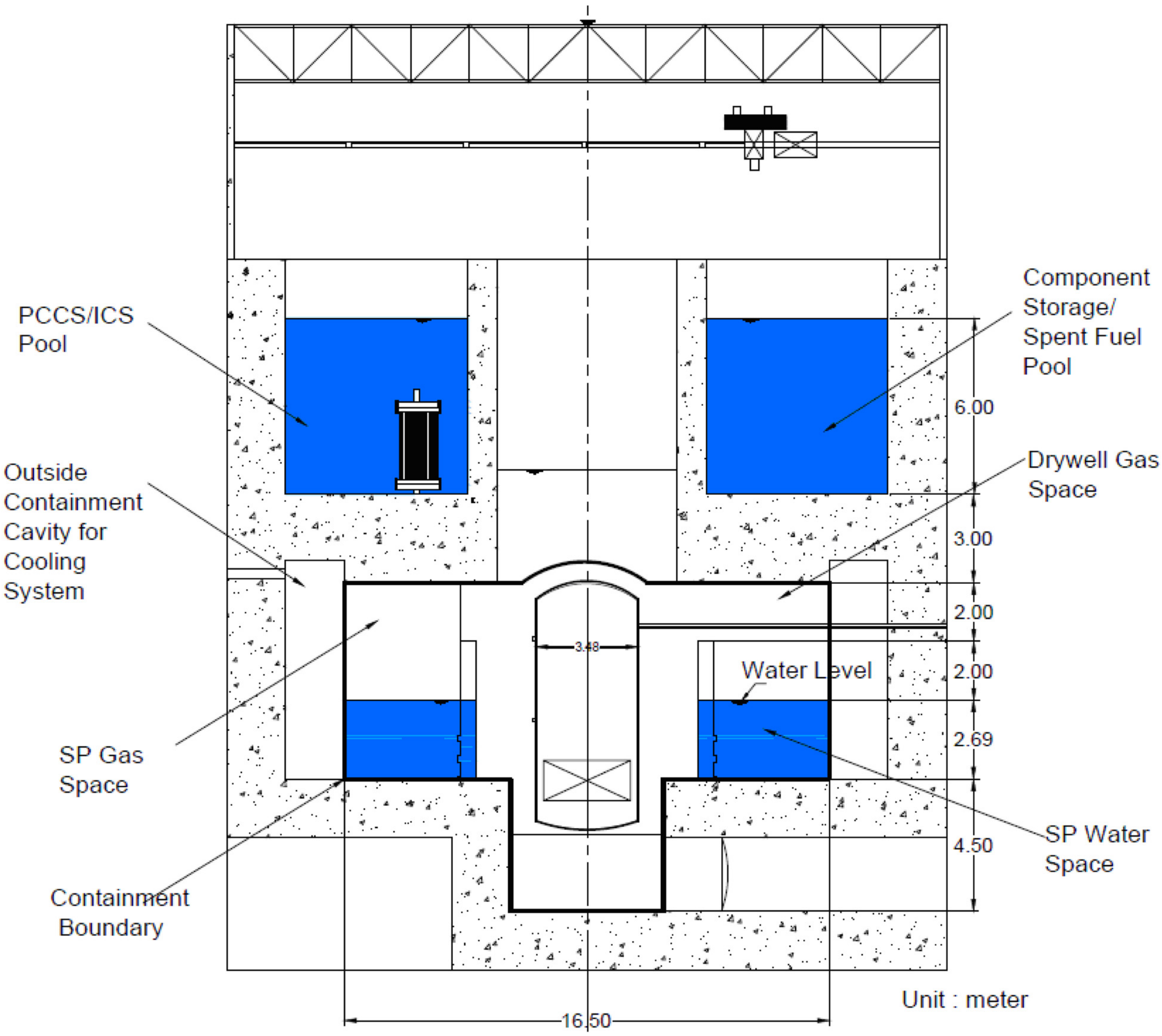

Fig. 1. Schematic view of passive safety systems of NMR-50 (Ref. 8).

TABLE II

Key Core Design Objectives and Constraints for NMR-50

Parameter	Value
Objectives	
Thermal power (MW)	165.0
Cycle length (years)	10.0
Constraints	
Maximum fuel enrichment (wt%)	5.0
Total power peaking factor	2.73
Axial power peaking factor	1.5
MFLPD (kW/m)	45.0
MCPR	1.32

was adopted since for a fixed assembly design, the cycle length decreases with increasing number of fuel batches while the discharge burnup increases with it. The single-batch fuel management scheme and the predetermined core configuration to maintain the natural-circulation capability with a reduced reactor vessel height readily render a core design as shown in Fig. 2 (due to geometric symmetry, only a quarter-core configuration is shown here).

As can be seen in Fig. 2, the NMR-50 core consists of 256 fuel assemblies in an 18 × 18 arrangement. Some important core parameters of NMR-50 are summarized in Table III. For comparison, the corresponding parameters of SBWR-600 and SBWR-200 are also presented in Table III. Note that the core diameter shown in Table III is the equivalent core diameter of all fuel assemblies including interassembly gaps in the core.

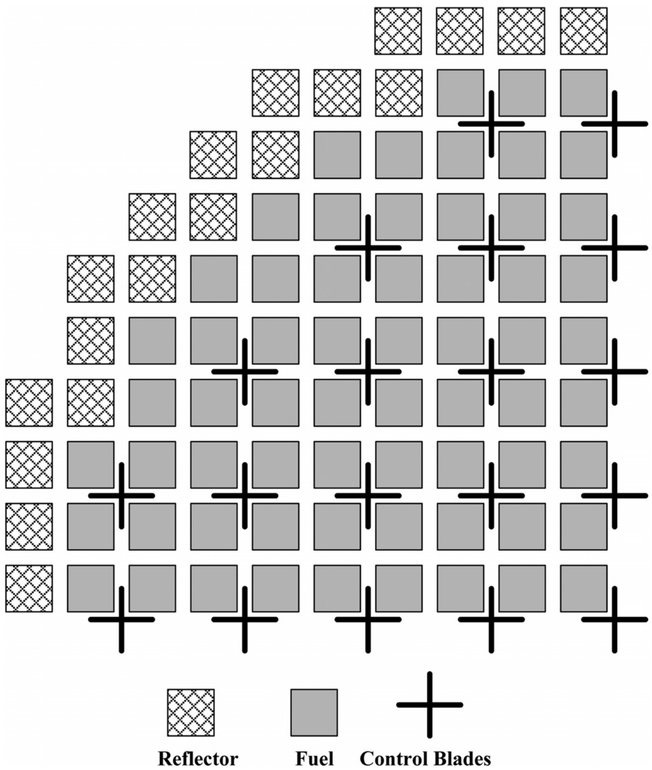

Fig. 2. Schematic view of the quarter-core of NMR-50.

With the above core configuration, the core design study was focused on the parametric studies to develop an optimum fuel assembly design that would have a sufficient excess reactivity to yield the targeted 10-year cycle length and the whole-core depletion analyses to confirm that the targeted cycle length would be achieved without violating the imposed design constraints on the reactivity coefficients and thermal limits. The parametric studies on fuel assembly design were performed using the CASMO-4 lattice physics code.[11] The whole-core depletion analyses with thermal feedbacks were carried out by performing coupled neutronics and T/H calculations using the PARCS core neutronics simulator[12] and the RELAP5 T/H analysis tool.[13]

Starting from the standard 8 × 8 fuel assembly design of GE used in SBWR-200 (Ref. 10) and the 10 × 10 ATRIUM 10B fuel assembly design of AREVA (Ref. 14), various fuel assembly designs were developed by varying the ^{235}U enrichment, fuel pin dimensions, number of gadolinium (Gd) burnable poison (BP) pins and BP enrichment, etc. The ^{235}U enrichment was limited to 5.0 wt% available in the current industry. The number of BP pins and their locations were determined to hold down the relatively high excess reactivity at the beginning of cycle (BOC) resulting from the single-batch fuel management scheme and to minimize the local power peaking within a fuel assembly. A few candidate assemblies were selected by carrying out assembly depletion calculations with CASMO-4 and estimating the cycle length based on the linear reactivity model.[15] For each of these candidate assembly designs, two-group cross sections were generated for full-core depletion analyses as a function of burnup, fuel temperature, moderator density, etc.

Using the two-group cross-section sets (in the format of the PMAXS file of PARCS) generated for each candidate assembly design, the core performance characteristics were evaluated using the PARCS code. Specifically, the excess reactivity, critical control blade position, power and burnup distributions, and power peaking factors were evaluated as a function of time. In order to account for thermal feedbacks, the PARCS neutronics calculations were coupled with the RELAP5 T/H calculations. The coupling calculations between PARCS and RELAP5 were carried out via a message transfer interface supported by a parallel virtual machine.[16] A schematic diagram illustrating the computational flow among different code systems is depicted in Fig. 3. The steady-state power and temperature distributions were iteratively determined by the PARCS and RELAP5 calculations, respectively.

To account for the thermal feedbacks through RELAP5 T/H calculations, fuel assemblies in the PARCS model were divided into three groups of similar power, and each group was represented by a single T/H channel. As a result, the NMR-50 core was modeled by four T/H

TABLE III

Comparison of Core Parameters of SBWR-600, SBWR-200, and NMR-50

Core Property	SBWR-600	SBWR-200	NMR-50
Thermal power [MW(thermal)]	2000	660	165
Number of fuel assemblies	732	256	256
Fuel assembly arrangement	30 × 30	18 × 18	18 × 18
Equivalent core diameter (m)	4.73	2.73	2.73
Active fuel length (m)	2.743	2.743	1.372
Number of control blades	177	57	57

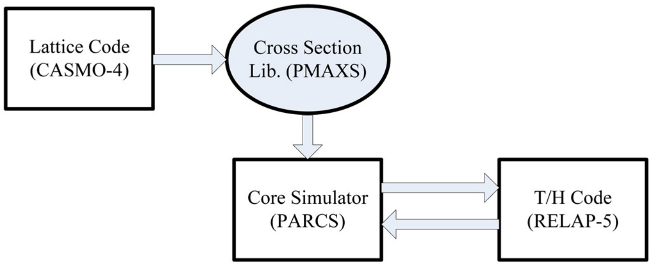

Fig. 3. Schematic view of the computational flow among nuclear analysis codes.

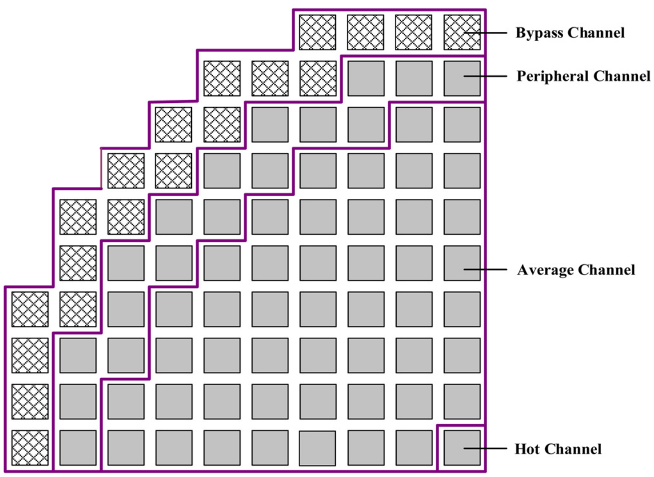

Fig. 4. Radial mapping of fuel assemblies to T/H channels.

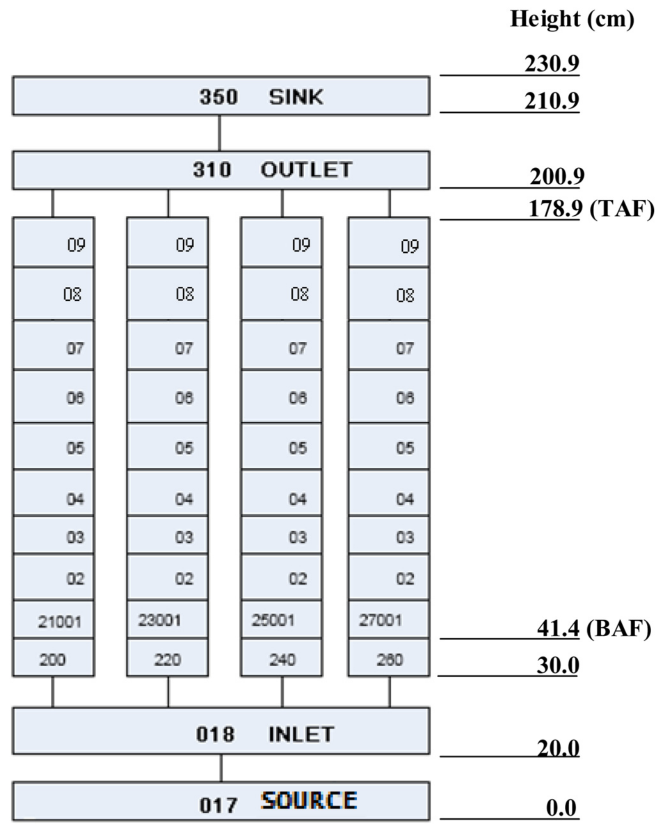

Fig. 5. RELAP5 nodalization diagram of NMR-50 core.

channels, including an additional bypass channel to account for the coolant flowing through the interassembly gaps as well as core reflector regions. Figure 4 shows the radial mapping of fuel assemblies to T/H channels, and Fig. 5 shows the RELAP5 nodalization diagram. As shown in Fig. 4, the fuel assemblies were divided into hot, average, and periphery channels. The hot channel represents the central 4 assemblies, the periphery channel represents the 68 assemblies at the core periphery, and the average channel represents the remaining 184 assemblies. The coolant flow in the core reflector regions and the interassembly gaps is represented by a separate bypass channel.

As shown in Fig. 5, for the RELAP5 calculations to account for thermal feedbacks, only part of the primary system was modeled with the boundary conditions derived from the complete primary system model analyses.[17] Each of the four channels was divided into ten axial nodes, and the four channels were connected at the inlet and outlet plena. Two additional nodes for coolant source and sink were included below the inlet plenum and above the outlet plenum, respectively. The inlet flow rate and temperature boundary conditions were imposed at the source node below the inlet plenum, and the outlet pressure boundary condition was imposed at the sink node above the outlet plenum. The boundary conditions used in the RELAP5 model as well as key T/H design parameters are shown in Table IV.

V. CORE DESIGN AND PERFORMANCE CHARACTERISTICS

Through the parametric studies discussed in Sec. IV, a fuel assembly design similar to AREVA's ATRIUM 10B design was developed for the NMR-50 core. The fuel rod configuration of the assembly is depicted in Fig. 6. The dimensions (at cold condition) for the assembly, as well as some fuel characteristics of the assembly, are summarized in Table V. As shown in Fig. 6, the fuel assembly consists of 91 fuel rods laid over a 10 × 10 grid with a square-shaped coolant channel in the center. Different fuel rod colors shown in Fig. 6 indicate different fissile enrichments in the fuel. Fuel rods with gadolinium BP are drawn with red color.

Each fuel pin has a 137.2-cm-long active fuel and a 15.24-cm-long graphite reflector at the top and bottom of the active fuel, respectively. The active fuel region of a Gd fuel rod is divided into two segments (of 45.75-cm and 91.45-cm height) with different amounts of Gd. The

TABLE IV

Key T/H Design Parameters of NMR-50 Core

Parameter	Value
Coolant flow rate (kg/h)	2.230×10^6
Core-averaged power density (kW/ℓ)	20.75
Nominal coolant outlet pressure (MPa)	7.178
Active fuel length (m)	1.372
Average coolant exit quality	0.143
Core-averaged coolant void fraction	0.455
Coolant saturation temperature (°C)	287.3
Coolant inlet temperature (°C)	278.5
Total flow area (m^2)	4.013
Bypass flow area (m^2)	1.763

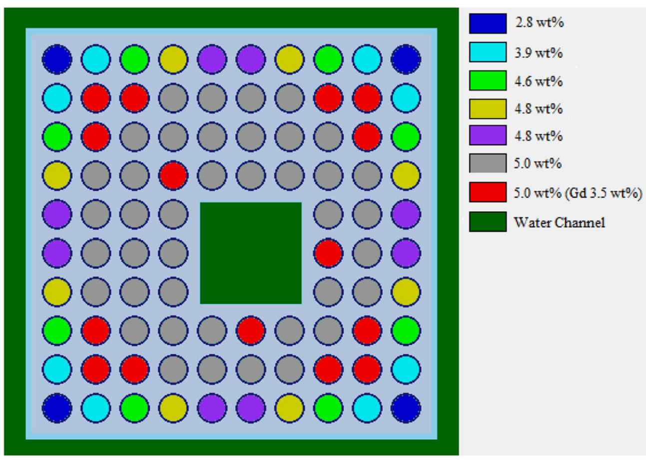

Fig. 6. A schematic view of the NMR-50 fuel assembly.

bottom segment contains 5 wt% of Gd, and the top segment contains 3.5 wt% of Gd. The basic strategy of this design is to counteract the reactivity penalty due to the coolant boiling in the upper regions of the core. The axial zoning of a Gd fuel rod is depicted in Fig. 7.

Using this fuel assembly design, full-core calculations were performed using the PARCS and RELAP-5 codes as discussed in Sec. IV. The multiplication factor k_{eff}, the critical control blade positions, the MFLPD, and the MCPR were calculated throughout the burn cycle. The criticality at each burn state was maintained by adjusting the axial positions of the 57 control blades, the layout of which is shown in Fig. 2. The critical control blade positions were determined in a way to reduce the total power peaking factor. As an example, the critical positions of control blades at BOC are shown in Fig. 8. The values in Fig. 8 denote the notch values of individual control blades: 3192 for a fully inserted control blade and zero for a fully withdrawn one. At BOC, the 15 Gd BP rods in each fuel assembly hold down the excess reactivity significantly, and thus, only 13

TABLE V

Design Parameters of NMR-50 Fuel Assembly

Parameter	Value
Fuel Characteristics	
Fuel density (g/cm^3)	10.45[a]
Average ^{235}U wt%	4.75
Average Gd wt% in Gd rod	4.00
Water/fuel volume ratio	2.33
Specific power (W/g U)	8.76
Local assembly peaking factor	1.27
k_{inf} at BOC	1.06059
Fuel Pin Dimensions	
Fuel rod outside diameter (mm)	10.55
Fuel rod cladding thickness (mm)	0.6055
Pellet-to-cladding gap (mm)	0.0851
Active fuel length (cm)	137.2
Fuel Assembly	
Number of fuel rods	91
Number of Gd fuel rods	15
Fuel pin pitch (mm)	12.95
Fuel assembly pitch (mm)	152.4

[a]The density for Gd-enriched fuel is slightly reduced.

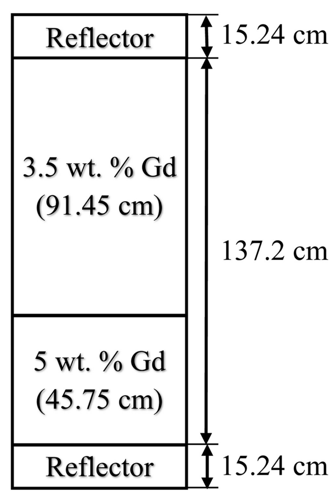

Fig. 7. Axial zoning of the gadolinium fuel rod.

			0	0
		0	0	0
	0	0	0	229
0	0	0	229	249
0	0	229	249	270

Fig. 8. Critical control blade positions at BOC.

central control blades are inserted slightly into the core as shown in Fig. 8.

The radial power distribution in the core at BOC is shown in Fig. 9, and the axial power distributions of three T/H channels are shown in Fig. 10. These power distributions represent the converged solution of the iterative neutronics and T/H calculations.

The resulting MFLPD and MCPR values at BOC of the NMR-50 design are compared in Table VI with those of the SBWR-600 design.[9] As can be deduced from the design parameters presented in Table III, the active fuel length of NMR-50 is about half of SBWR-600, and the NMR-50 fuel assembly has 91 fuel pins whereas the SBWR-600 fuel assembly has 60 fuel pins (GE's standard 8 × 8 fuel assembly design). As a result, the average linear power density of NMR-50 is lower by a factor of 3.2 than that of SBWR-600. Since the total power peaking factor of the NMR-50 core is slightly higher than that of the SBWR-600 core, the MFLPD of NMR-50 is only one-third of the corresponding value of SBWR-600. This provides a significantly larger margin to the limiting peak cladding temperature than the SBWR-600 design. The MCPR value was estimated by iteratively determining the axial position for the onset of the saturated boiling point corresponding to the critical power using the Hench-Gillis correlation,[18] which represents the critical quality of a heat transfer channel under consideration as a function of the boiling length of the channel. The estimated MCPR of NMR-50 at BOC is ~2.25, which is significantly higher than the operation limiting value of 1.32 shown in Table II.

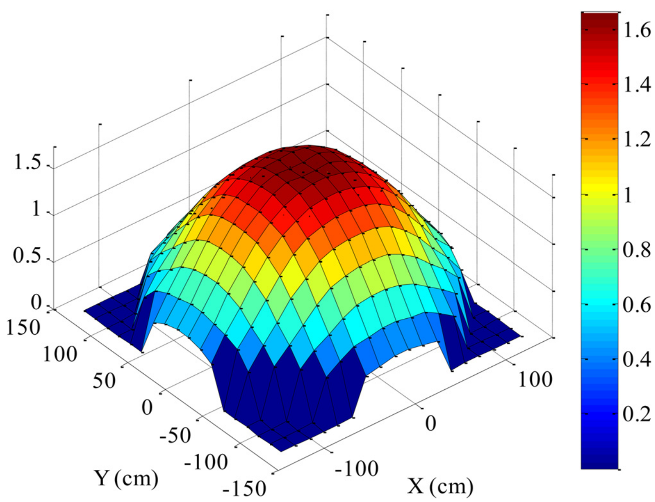

Fig. 9. Radial power distribution of NMR-50 at BOC.

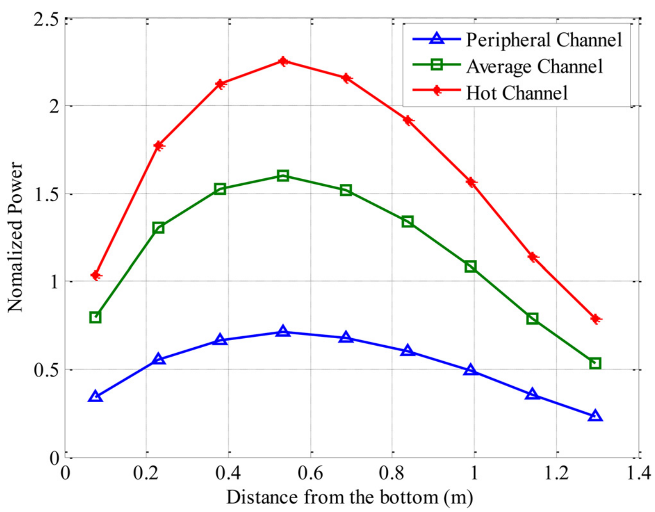

Fig. 10. Axial power distribution at BOC for different flow channels.

TABLE VI

The T/H Properties of NMR-50 at BOC

Property	SBWR-600	NMR-50
Average linear power density (kW/m)	16.60	5.16
Total power peaking factor	2.73	2.98
MFLPD (kW/m)	45.30	15.36
MCPR	1.32	2.25

Table VII presents the multiplication factor k_{eff}, the sum of control blade notch values, and the MFLPD and MCPR values at selected burn times, and Fig. 11 shows the core-averaged axial power distributions at the BOC, the middle of cycle (MOC), and the end of cycle (EOC). The results in Table VII indicate that the targeted 10-year cycle length can be achieved with an average ^{235}U enrichment of 4.75 wt%. In fact, the inserted amount of control blades at ~10 years suggested that a cycle length longer than 10 years could be achieved. In addition, the estimated MFLPD and MCPR meet the imposed design constraints with significant margins throughout the burn cycle. As can be seen in Fig. 11, the axial power shape shows significant variations with burnup because of the movement of control blades and the depletion of BPs. The axial peaking factors for BOC, MOC, and EOC are 1.37, 1.40, and 1.65, respectively. Though the axial peaking factor at EOC is slightly higher than the selected limit given in Table II, it is acceptable since the MFLPD of NMR-50 is far below the limiting value of 45.0 kW/m. Moreover, the axial peaking factor can be reduced with further optimization of the fuel assembly design and the sequence of control blade movement. All these results clearly indicate the developed NMR-50 core design achieves the targeted 10-year cycle length while satisfying the thermal design constraints.

TABLE VII

Core Performance of NMR-50 in 10-Year Fuel Cycle Calculation

Burn Time (years)	Average Burnup (GWd/tonne)	k_{eff}	Control Blade Notch[a]	MFLPD (kW/m)	MCPR
0.00	0.00	0.99988	1 455	15.36	2.25
1.00	3.06	1.00560	14 394	17.78	2.55
2.00	6.12	1.00135	28 101	17.61	2.36
3.00	9.18	1.00062	40 818	18.66	2.17
4.00	12.24	1.00005	38 856	13.13	2.29
5.00	15.31	1.00010	34 602	12.48	2.47
6.00	18.37	1.00009	27 262	12.92	2.07
7.00	21.43	1.00009	23 346	11.97	2.34
8.00	24.49	1.00010	19 139	12.39	2.57
9.00	27.55	1.00011	14 490	14.06	2.84
9.99	30.61	1.00010	7 963	15.80	2.79

[a]The notch value is the sum of notches for all inserted control blades.

VI. PRELIMINARY SAFETY ANALYSES

Preliminary safety analyses were performed for the NMR-50 design using the point kinetics model and a RELAP5 model for the entire primary system. The fuel temperature and coolant void coefficients were calculated by varying the power level in the PARCS model and determining the corresponding changes in the core-averaged fuel temperature and coolant void fraction as well as the core reactivity. The core-averaged fuel and coolant temperatures at the normal operating condition are ~790 and 560 K, respectively, and the core-averaged coolant void fraction is ~45.5% (see Table IV). The resulting fuel temperature coefficient is −3.29 pcm/°C, and the void coefficient as a function of the void percentage is shown in Fig. 12. As shown in Fig. 12, the coolant void coefficient becomes more negative with increasing coolant void fraction, and its value at the operating condition is about −98 pcm/% void. The delayed neutron fractions and the neutron generation time were also evaluated as shown in Table VIII.

Two representative design-basis LOCAs were analyzed using the RELAP5 code: main steam line break (MSLB) and bottom drain line break (BDLB). For computation convenience, a one-dimensional process is used in the RELAP5 model with several independent safety systems lumped into one loop. The one-dimensional NMR-50 model is separated into several distinct sections, such as the RPV, dry well, wet well, ICS, and PCCS. In the accidental simulation, the reactor is initially assumed to be at full power and normal operation condition, and then the break is initiated. The scenario followed the progression of the accident including reactor scram and ADS actuation. A detailed discussion of

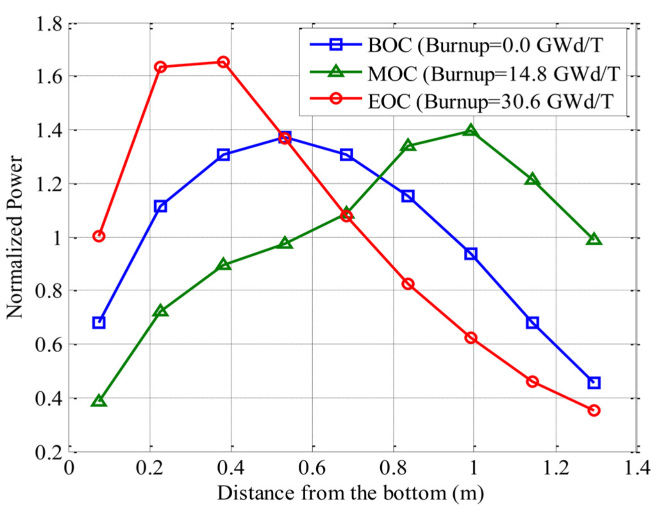

Fig. 11. Core-averaged axial power shapes at BOC, MOC, and EOC.

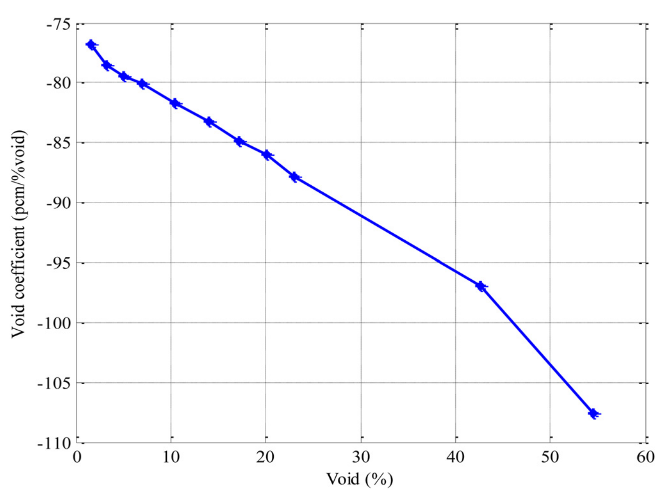

Fig. 12. The negative void coefficient curve for NMR-50.

TABLE VIII

Point Kinetics Parameters for NMR-50 at BOC

Group	1	2	3	4	5	6
λ_i (s)[a] β_i[b]	0.012754 0.000241	0.031784 0.001429	0.119534 0.001313	0.319024 0.002889	1.402641 0.001037	3.930236 0.000247
$\beta = \Sigma \beta_i$[c] Λ (s)[d]	0.007156 0.000022					

[a]λ_i = Average decay constant of group *i*.
[b]β_i = Delayed neutron fraction of group *i*.
[c]β = Effective delayed neutron fraction.
[d]Λ = Average neutron generation time.

the postulated accidents can be found in a related paper.[17] As a proof of principle, key results of the two accidents are presented here to demonstrate the passive safety characteristics of NMR-50. The collapsed water levels in the RPV change along with the time after the initiation of MSLB and BDLB are shown in Figs. 13 and 14, respectively.

As shown in Figs. 13 and 14, in either accident, the water level dropped quickly after the break and then increased due to the opening of the SP equalization lines (EQL) and PCCS drain flow back to RPV. As can be seen, the collapsed water level in the RPV, in both accidents, was always above the top of active fuel (TAF). Since the two-phase mixed water level is usually significantly higher than the collapsed level, it is safe to state that the core will remain covered during the entire transient procedure of the two accidents.

VII. SUMMARY AND CONCLUSIONS

Core design studies were performed to develop a NMR-50 core design to yield a 10-year cycle length with fuel enrichment within the industrial limit of 5 wt% while satisfying other operation and safety-related design constraints. Assembly design studies and the generation of two-group cross sections for whole-core calculations were performed using the CASMO-4 lattice code. The whole-core calculations were carried out using the PARCS neutronics simulator and the RELAP5 T/H code.

A 10 × 10 fuel assembly design was developed starting from AREVA's ATRIUM 10B design, and a NMR-50 core design was developed using a single-batch fuel management scheme. The normal operation status of the NMR-50 core was determined iteratively by coupled calculations of PARCS and RELAP5. The resulting MFLPD at BOC was 15.4 kW/m, being far below the imposed design limit of 45 kW/m. The evaluated MCPR based on the Hench-Gillis correlation was 2.25, which meets the operation design limit of 1.32 adopted from the SBWR-600 design. The core depletion results showed that the developed NMR-50 core design achieves the targeted 10-year fuel cycle length while satisfying the MFLPD and MCPR limits with significant margins through the 10-year

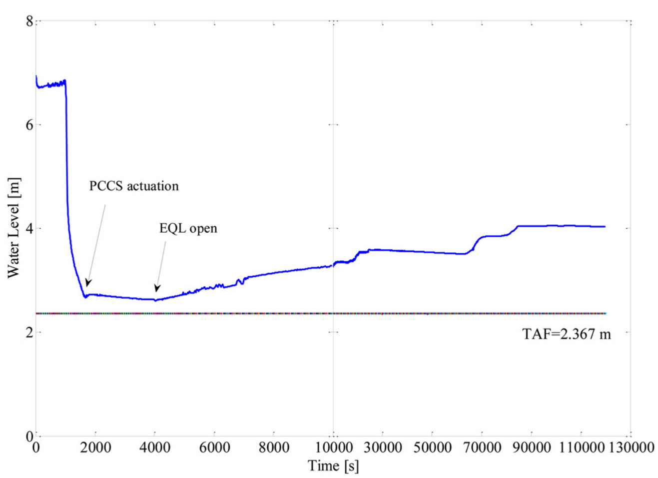

Fig. 13. Collapsed water levels in the RPV (MSLB) (Ref. 17).

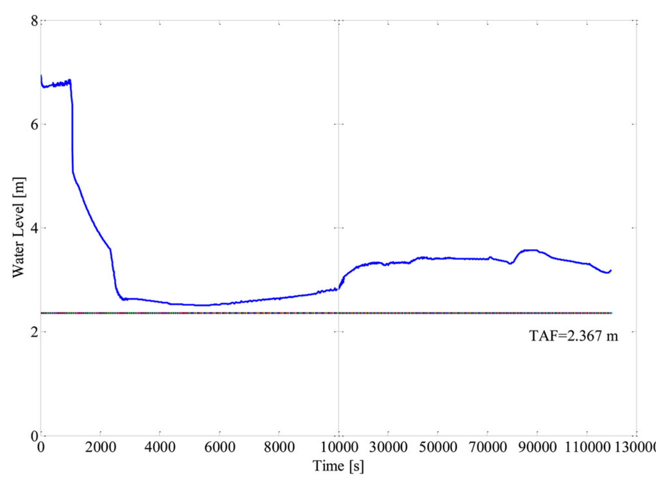

Fig. 14. Collapsed water levels in the RPV (BDLB) (Ref. 17).

burn cycle. Preliminary safety analyses showed that the core would remain covered during the entire transients of the MSLB and BDLB design-basis LOCAs. These results indicate that the targeted 10-year cycle length is achievable while satisfying the operation and safety-related design criteria with sufficient margins.

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