Research Article

New Methods for Evaluation of Spent Fuel Condition during Long-Term Storage in Slovakia

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Experiences with an advanced spent nuclear fuel management in Slovakia are presented in this paper. The evaluation and monitoring procedures are based on practices at the Slovak wet interim spent fuel storage facility in NPP Jaslovské Bohunice. Since 1999, leak testing of WWER-440 fuel assemblies are provided by special leak tightness detection system “Sipping in pool” delivered by Framatomeanp with external heating for the precise defects determination. In 2006, a new inspection stand “SVYP-440” for monitoring of spent nuclear fuel condition was inserted. This stand has the possibility to open WWER-440 fuel assemblies and examine fuel elements. Optimal ways of spent fuel disposal and monitoring of nuclear fuel condition were designed. With appropriate approach of conservativeness, new factor for specifying spent fuel leak tightness is introduced in the paper. By using computer simulations (based on SCALE 4.4a code) for fission products creation and measurements by system “Sipping in pool,” the limit values of leak tightness were established.

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1. Introduction

The Interim Spent Fuel Storage Facility (ISFSF) in Jaslovské Bohunice [1–3] is an important component of the spent nuclear fuel management system. The facility has been used for storage purposes since 1987. ISFSF is a nuclear facility providing for a safe storage of the spent nuclear fuel from VVER-440 reactors for the time period of 50 years before the fuel is further processed in a reprocessing plant or appropriately disposed off.

It is necessary to keep the concentration of fission products in storage pools on the low level for assurance of acceptable activity of the coolant. This can be done with periodical monitoring of the fuel elements condition, defects identification, and closing of leaking assemblies or fuel elements, respectively, in special hermetic caskets. This was the main reason for including not only “Sipping in pool” system, but also inspection stand “SVYP-440” (Figure 1), into the ISFSF operation [4–7].

The system “Sipping in pool” was built and implemented in the storage facility operation in 1999 and since then,

the important results have been measured. The system increases the temperature of the fuel assembly (by external heaters), which cause the increasing of the pressure inside fuel elements. If there is any leakage, increased pressure will cause higher fission product release. By measurement of released activity, the assembly tightness is determined.

Since December 2006, the new stand for VVER-440 fuel assemblies’ inspection “SVYP-440” is in operation. By using several modules, it has ability to open and take the fuel assembly apart, so it can examine all fuel elements. If the defect is found, fuel element with defect is closed into the special hermetic case.

2. Experimental

Slovakia has more than 20 years of experience with spent fuel storage. Since beginning, there were no leakages detected during storage conditions. Even though the negative effects of fuel cladding are very low, however, due to degradation of Zirconium alloys after long periods of under water storage, there is a finite possibility of defect formation. It is also

FIGURE 1: Equipment for VVER-440 spent fuel assemblies control in Slovakia.

difficult to estimate the long-term degradation process. With use of the systems like “Sipping in Pool” or “SVYP-440,” the leakages of the fuel assembly can be detected.

In that case, it is necessary to have as conservative approach to the extent possible. If we compare the volume activity of released fission products around fuel assembly A_O (measured by “Sipping in Pool”) and the volume activity of fission products inside whole fuel assembly A_I (calculated by SCALE code), we could estimate the fuel cladding condition. Therefore, we are introducing the *fuel cladding leak tightness coefficient* k_{FCT} :

$$k_{FCT} = \frac{A_O}{A_I}. \quad (1)$$

For the calculations of volume activity of fission products (in particular, ^{137}Cs has been considered) inside whole fuel assembly, sequence ORIGEN-ARP have been used (for version SCALE 4.4a) [8, 9]. As a simplification of the calculations, one model of fuel assembly has been used for all measured assemblies. This model, created by VUJE, a.s. company (Ing. Vladimír Chrapčiak, Ing. Radoslav Zajac, December 2002), is a standard model of VVER-440 fuel assembly, with the UO_x fuel and 4.2% enrichment of ^{235}U . Only burnup, power, effective days during operation and days during shut downs were unique for every assembly.

3. Results

After SCALE calculations of volume activity of ^{137}Cs inside the fuel assemblies, and measurements of volume activity of ^{137}Cs outside the fuel assemblies by “Sipping in Pool,” 36 values of the fuel cladding leak tightness coefficients k_{FCT} have been obtained (Table 1). All values were described by distribution function (Figure 2) to determine the mean value μ and standard deviation σ . It means that all fuel assemblies with $k_{FCT} = 1.1 \cdot 10^{-10} \pm 6.5 \cdot 10^{-11}$, or $(\mu + 1 \cdot \sigma)$,

FIGURE 2: Distribution function of continuous variable k_{FCT} .

are without any leakages. Fuel assemblies’ producer criteria for the released γ activity of fission products are 10^{-4} Ci/L ($3.7 \cdot 10^6 \text{ Bq/L}$). Because there were only small differences (less the 10^1) between the measured summary gamma activity and the separated cesium activity, we compared these two activities.

There were two limit values calculated:

$$(a) \ k_{FCT} = 3 \cdot 10^{-10},$$

$$(b) \ k_{FCT} = 8 \cdot 10^{-7}.$$

The first value, $k_{FCT} = 1.1 \cdot 10^{-10} \pm 1.95 \cdot 10^{-10} = 3 \cdot 10^{-10}$, or $(\mu + 3 \cdot \sigma)$ is from the statistical dispersion of distribution function of continuous k_{FCT} , and means that 99.73% of the values are within 3 standard deviation. In other words, $k_{FCT} = 3 \cdot 10^{-10}$ means that with the probability of 99.73% will be all measured and calculated assemblies tight.

TABLE 1: Measured and calculated values of A_O and A_I and final calculation of k_{FCT} .

Number of FA	A_O [Bq/L]	A_O [Bq]	A_I to 1 kgU [Ci]	A_I to 1 FA [Bq]	k_{FCT}
1	639	63 900	82.7	3.678E+14	1.73736E-10
2	609	60 900	82.7	3.678E+14	1.65579E-10
3	527	52 700	115	5.11451E+14	1.0304E-10
4	563	56 300	113	5.02556E+14	1.12027E-10
5	535	53 500	113	5.02556E+14	1.06456E-10
6	609	60 900	118	5.24793E+14	1.16046E-10
7	421	42 100	86.4	3.84255E+14	1.09563E-10
8	584	58 400	81.3	3.61574E+14	1.61516E-10
9	483	48 300	102	4.53635E+14	1.06473E-10
10	539	53 900	86.4	3.84255E+14	1.40271E-10
11	484	48 400	80.8	3.5935E+14	1.34688E-10
12	536	53 600	80.8	3.5935E+14	1.49158E-10
13	627	62 700	100	4.4474E+14	1.40981E-10
14	135 000	13500 000	103	4.58082E+14	2.94707E-08
15	6740	674 000	109	4.84767E+14	1.39036E-09
16	74 000	7400 000	109	4.84767E+14	1.52651E-08
17	372	37 200	79	3.51345E+14	1.05879E-10
18	366	36 600	79	3.51345E+14	1.04171E-10
19	277	27 700	122	5.42583E+14	5.10521E-11
20	441	44 100	122	5.42583E+14	8.12779E-11
21	461	46 100	125	5.55925E+14	8.29249E-11
22	288	28 800	132	5.87057E+14	4.90583E-11
23	437	43 700	77.7	3.45563E+14	1.2646E-10
24	310	31 000	77.7	3.45563E+14	8.97087E-11
25	505	50 500	111	4.93661E+14	1.02297E-10
26	346	34 600	111	4.93661E+14	7.00885E-11
27	854	85 400	114	5.07004E+14	1.68441E-10
28	238	23 800	114	5.07004E+14	4.69425E-11
29	320	32 000	75.6	3.36223E+14	9.51748E-11
30	182	18 200	75.6	3.36223E+14	5.41307E-11
31	278	27 800	126	5.60372E+14	4.96099E-11
32	150	15 000	118	5.24793E+14	2.85827E-11
33	366	36 600	119	5.29241E+14	6.91557E-11
34	105	10 500	124	5.51478E+14	1.90398E-11
35	2030	203 000	74.5	3.31331E+14	6.1268E-10
36	3880	388 000	74.5	3.31331E+14	1.17103E-09

The second value is calculated with use of fuel assemblies' producer criteria $A_{OP} = 3.7 \cdot 10^6$ Bq/L

$$k_{FCT} = \frac{A_O}{A_I} = \frac{3.7 \cdot 10^8 \text{ Bq}}{4.4655 \cdot 10^{14} \text{ Bq}} = 8.06 \cdot 10^{-7}, \quad (2)$$

where $A_O = A_{OP}^* V_s$, V_s : volume of the Sipping casket [3, 4], and A_I : average value of the calculated A_I (Table 1).

By considering by this value, we set the final limit intervals:

- (1) $(0 - 3 \cdot 10^{-10})$: values of the fuel cladding leak tightness coefficient for tight fuel assembly— $k_{FCT}(T)$,

- (2) $(3 \cdot 10^{-10} - 8 \cdot 10^{-7})$: values of the fuel cladding leak tightness coefficient for fuel assembly suspicious from leakage— $k_{FCT}(SL)$,

- (3) $(8 \cdot 10^{-7} - 1)$: values of the fuel cladding leak tightness coefficient for fuel assembly with leakage— $k_{FCT}(L)$.

The optimal use of the cladding leak tightness coefficient k_{FCT} is its application during periodical monitoring of fuel assemblies. Because of the low level of summary γ activity in the coolant, monitoring of 6 fuel assemblies at ISFSF in Jaslovské Bohunice is provided once per year. Two assemblies are marked as reference, other 4 assemblies are chosen randomly. For those two reference assemblies,

FIGURE 3: Limit values for the fuel cladding leak tightness coefficient k_{FCT} .

FIGURE 4: Values of k_{FCT} for reference FA number 1.

values of the cladding leak tightness coefficient k_{FCT} were calculated. For both assemblies (Figures 4 and 5), the values of k_{FCT} are about $1.2 \cdot 10^{-10}$. Only during last measurement, few deviations have been observed. The deviation for first assembly is $+7.73 \cdot \sigma$, and for second assembly it is $+16.3 \cdot \sigma$. This means, that both fuel assemblies are susceptible to the leak tightness. However, during measurements in 2007, values of k_{FCT} increased for every measured assembly. So, the “jumps” can be caused by incorrect measurements by “Sipping in Pool.”

4. Conclusions

From the nuclear safety point of view, it is necessary to keep the fission products inside the fuel elements and to prevent their escape into environment not only during reactor operation or fuel transport, but also during the long-term storage of spent nuclear fuel. Therefore, the effective leak tightness monitoring system at all fuel interim storages is necessary. The designed system from the 80s at the Slovak wet interim storage facility did not assure this task at the desired level, so the system “Sipping in Pool” was implemented in 1999. After several years of its operation, performed measurements showed that this system is highly effective equipment for fuel cladding defects detection. Since 2006 a new inspection stand “SVYP-440” for monitoring of spent nuclear fuel condition is used as well.

FIGURE 5: Values of k_{FCT} for reference FA number 2.

New factor for specifying of spent fuel leak tightness has been introduced. Using the computer simulations (based on SCALE 4.4a code) for fission products creation and measurements by the system “Sipping in Pool,” the limit values of the cladding leak tightness coefficient k_{FCT} has been defined. It is a tool, which is used as additional information, describing fuel cladding leak tightness criteria. Forward-looking, the authors expect that the mean value of k_{FCT} will oscillate about 10^{-10} . Contingent deviations could be caused by incorrect measurements by “Sipping in Pool,” or using of incorrect fuel assembly model for SCALE calculations. Also, the leak tightness coefficient k_{FCT} will depend on the pool water cleaning system. Depending on the residual activity in the pool, the values of k_{FCT} will change. Therefore, further research is needed.

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