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Preliminary Evaluation of the Source Terms for Nuclear Fuel Irradiation Tests in HANARO

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1. Introduction

An in-pile test program for the development of a high burn-up fuel is planned for the HANARO reactor. Three test fuel pins having 5 UO_2 pellets each are being loaded into the capsule for the tests. If the fuel is not intact, even if operated in a closed system, radioactive gases will be released to the gas flow tube from the instrumented capsule and the I&C system (GSF-2002) and could also affect the health of the reactor workers and the public. For this reason, a series of analyses is being studied in 2005 and it has three main objectives; namely an attempt to understand the total fission gas release from the test fuel pins, a potential safety hazard at the HANARO reactor, and the required shielding for mitigating the risk from a radiation to reactor workers.

2. Methods and Results

2.1 ORIGEN-code validation

To investigate the applicability of the ORIGEN2.2 code [1] to the HANARO reactor study is one of the major purposes of this study. The ORIGEN-2 computer code does not provide exact activity data of the fuels irradiated in the HANARO reactor because it has several specific libraries such PWR, BWR, etc. However, a trending of the produced nuclides can be of benefit in evaluating various aspects of a fuel performance. For the validation of the ORIGEN2 code, the activities of the total fission products, especially Kr-85, released from PWR fuels corresponding to a burn-up of 41.8 and 55 GWD/MTU are calculated.

The predicted activity of Kr-85 calculated by the code was compared with the PIE data which was obtained from an accurate radiochemical measurement for UO_2 fuels. The difference between each calculated and measured data as given in Table 1 have very similar levels of agreement, indicating that the code works quite well over a wide range of burn-ups.

Table 1. Comparison of measured and calculated Kr-85 activity

Fuel Type(GWd/tu)	Measured Average Data(Ci)	Calculated Data by ORIGEN-code(Ci)
Young Kwang 1 (55)	2.870	1.64
Kori 2 (41.8)	0.131	0.091

2.2. Test fuel characteristics and fission gas

The test fuel was short rods of Zr-4 clad UO_2 pellets of a typical PWR 17x17design. As used in pressurized water reactors, a UO_2 pellet is a small cylinder approximately 8.19mm in diameter and 10.2mm in length. The density, average grain size and enrichments of the test fuel pellet provided from Korea Nuclear Fuel Fabrication Co are 95.8%T.D., 9.35µm and 2.42 percent uranium-235, respectively.[2, 3] The irradiation will be done in an instrumented capsule for a total of 4 cycles (one cycle lasts for 21 days with 30 MWt). The thermal flux is in the range 3 to 4 x 10^{18} $n/m^2.s$ for a total time of 2016 h giving a cumulative burn-up of around 6 GWD/MTU.

The total fission products produced in the test fuel pellets being irradiated in the HANARO reactor are analyzed by the ORIGEN-2 computer code. The analyzed volatile nuclides for a given burn-up are given in Table 2. The noble gases and the iodines among the groups are the important volatile nuclides we are concerned with.

Table 2. Cumulative gaseous fission products for 17 weeks

Nuclides	21 Days	119 Days	Nuclides	21 Days	119 Days
KR 83M	2.28E+01	2.16E+01	I128		6.36E-02
KR 85	4.30E-02	1.67E-01	I129		3.26E-07
KR 85M	5.34E+01	4.99E+01	I130		3.00E-01
KR 87	1.08E+02	9.99E+01	I131	1.05E+02	1.10E+02
KR 88	1.52E+02	1.41E+02	I132	1.86E+02	1.87E+02
XE131M	6.08E-01	1.69E+01	I133	2.89E+02	2.86E+02
XE133	2.58E+02	2.56E+02	I134	3.26E+02	3.20E+02
XE133M	8.03E+00	8.07E+00	I135	2.69E+02	2.66E+02
XE135	9.12E+01	7.94E+01			

2.3. Inert fission gas and iodine release

To calculate the release to birth ratio(R/B) of the fission gas of the fuel, diffusion coefficients by Friskney and Turnbull [4] are used. They also measured the release rates for the unstable rare gases of Kr-88, Kr-87, Kr-87m, Xe-133, Xe-135, Xe-138 for samples of polycrystalline and large-grained uranium dioxide in the temperature range 700-1550 °C. The calculated release rates [4, 5] are given in Table 3.

The applicable regulations at HANARO require the fuel to be intact during a reactor operation. A fuel cladding failure can be caused by manufacturing defects. For the typical PWR fuel the cladding is generally a vital barrier between radioactive materials and the environment. Thus, from a safety point of view, potential safety hazards at the HANARO reactor which operates with small cracks and holes in the cladding, containing their fuel are reviewed. As additional metal cladding was adopted in our case, the fuel cladding is not a vital barrier. Even if it is not intact, fission product gases retained in the test fuel pins are released to the gap between the fuel cladding and the mini-capsule tube, where radiation monitors quickly identify fission products released from the fuel cladding failure in each mini-capsule separately mounted inside the basket. Such a release could affect the health of radiation workers at the HANARO reactor.

Table 3. R/B ratio with the temperature

T(℃)	10^4/T(K)	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-135	Xe-138	I-131	I-133
1451	5.8	1.45-2	1.1-2	8.8-3	2.1-1	4.5-2	2.5-3		
1290	6.4	8.7-3	5.6-3	4.6-3	8.6-2	1.7-2	1.35-3	0.42	3.43-2
1156	7.0	5.2-3	3.0-3	2.6-3	4.5-2	7.3-3	8.2-4	6.29-2	2.34-2
1043	7.6	3.3-3	1.8-3	1.5-3	3.1-2	3.8-3	5.2-4		
947	8.2	2.6-3	1.3-3	1.2-3	2.4-2	2.6-3	3.7-4	1.39-2	9.41-3
890	8.6	2.3-3	1.2-3	1.05-3	2.0-2	2.4-3	3.2-4		
863	8.8	2.15-3	1.2-3	1.02-3	1.9-2	2.25-3	3.0-4		
814	9.2	1.9-3	1.1-3	9.4-4	1.65-2	2.1-3	2.75-4	1.07-2	5.97-3
769	9.6	1.75-3	1.0-3	8.6-4	1.4-2	1.9-3	2.6-4		

(Note: $1.23\times10^{-3} \rightarrow 1.23\text{-}3$)

The magnitude and composition of the released fission products depends on the size of the defect and the number of defective fuel rods in the capsule. The activity leakage rate was reported by Page [6]. The leakage rate was derived from the measured activity concentration in the coolant of the PLUTO reactor as given in the Table 4.

Table 4. Fission gas leakage from the fuel pin [6]

Nuclides	Leakage(%/10min.)	Nuclides	Leakage(%/10min.)
Kr-83m	3	Xe-135m	11
Kr-85m	<1	I-131	3
Kr-88	14	I-132	3
Xe-131	~	I-133	13
Xe-133	~	I-134	3
Xe-133m	~	I-135	5
Xe-135	10		

2.4 Source term generation

The source terms from the three test fuel pins are evaluated using the uranium mass, burn-up level, R/B ratio, and leakage rates of the fission gases. Short lived source nuclides are not included in this analysis for the calculation of the exact mixed source. Table 5 shows the fission gases released from the test fuel pins for 1 minute after the cladding failure. Even though the technical bases for estimating the release rate is similar, it appears likely that a comparison of the data to similar cases is not easy now. The source terms will be utilized to design the shielding for the instrumented capsule which is very important to radiation safety. Micro-shield code, one of the current popular shielding codes, will be used. The purpose of the work is to confirm that the allowable dose limit on the surface of the I&C system including the gas flow tube is met. If excessively high dose rates are expected, corrective actions can be made to ensure that the radiation sources are less than the allowable limits.

Table 5. Fission gas leakage from the fuel pin for 1 minute

Nuclide	Burning during 21 days(Ci)	Release to birth ratio(890℃) (small grain size)	Assumption of R/B	Total release (Ci)	Leakage ratio (%)/min	Total leakage for 1 min.(Ci)
KR 83M			3.0E-03	0.06843	0.3	0.00020529
KR 85	1.67E-01		3.0E-03	0.000128	0.1	1.2897E-07
KR 85M	4.99E+01	2.30E-03		0.122843	0.1	0.00012284
KR 87	9.99E+01	1.20E-03		0.12936	1.4	0.00181104
KR 88	1.41E+02	1.05E-03		0.15981	1.4	0.00223734
XE131M	1.69E+01		2.0E-02	0.012164	0	0
XE133	2.56E+02	2.00E-02		5.15	0	0
XE133M	8.07E+00		2.0E-02	0.16054	0	0
XE135	7.94E+01	2.40E-03		0.218808	1	0.00218808
I131	1.10E+02	0.01070		1.12243	0.3	0.00336729
I132	1.87E+02		0.01	1.856	0.3	0.005568
I133	2.86E+02	0.00597		1.72533	1.3	0.02242929
I134	3.20E+02		0.01	3.258	0.3	0.009774
I135	2.66E+02		0.01	2.692	0.5	0.01346

3. Conclusion

From the preliminary evaluation of the source terms the following results are obtained:

- From the code validation work for the activity calculation of the total fission products released from the PWR type spent fuels, it was confirmed that the ORIGIN 2.2 code works quite well.
- Based on the data of fission products nuclides, release to birth ratio with the temperature, and the leakage rate per unit time from the test fuel pin being irradiated in HANARO are made successfully.
- From the calculation of the mixed source, details of the produced isotopes are made for the shielding analysis.

This data will be used for the review of a potential safety hazard through a detailed shielding analysis.

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References

[1] S. B. Ludwig and J. P. Renier "Standard- and Extended-Burn-up PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018, December 1989.

[2] Y.H. Kang, et al., “Research Reactor Utilization Tech. Development”, KAERI/RR-2360/03, KAERI, 2003.

[3] B.G. Kim, et al, “Irradiation Test Plan of an Instrumented Capsule for Nuclear Fuel Irradiation in HANARO”, KAERI/TR-2774/ 2004, KAERI, 2004.

[4] C.A. Friskney and J.A. Turnbull, J. Nuclear Materials, 79(1979) 184-198.

[5] Ma. Benjamin M., Nuclear Reactor Materials and Applications, Van Nostrand Reinhold Company Inc, New York (1983) pp 162-182.

[6] J.D page, RWE Solutions, KIFC/DN03, MD31481 (2001)1-11.