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Effect of Co-free Valve on Activity Reduction in PWR

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ABSTRACTS

Radioactive nuclei, such as Co^{58} and Co^{60}, deposited on out-of-core surfaces in a pressurized water reactor (PWR) primary coolant system, are major sources of occupational radiation exposure to plant maintenance personnel and act as costly impediment to prompt and effective repairs. Valve hardfacing alloys exposed to primary coolant are considered as one of the main Co sources. To evaluate the Co-free valve, such as NOREM 02 and Deloro 50, the candidates for the alternative to Stellite 6, in a simulated PWR primary condition, SNU COrrosion Test Loop (SCOTL) was constructed. For gate valves hardfaced with made of NOREM 02 and Deloro 50 hot cycling tests were conducted for up to 2,000 on-off cycles with cold leak tests at 1,000 cycle interval. It was observed that the leak rate of NOREM 02 (Fe-base) did not satisfy the nuclear grade valve leak criteria. After 1000 cycles test, while there was no leakage in case of Deloro 50 (Ni-base). Also, Deloro 50 showed no leakage after 2000 cycles. To estimate the activity reduction effect, we modified CRUDSIM-MIT which modeled the effects of coolant chemistry on the crud transport and activity buildup in the primary system of PWR. In the new code, CRud Evaluation and AssessmenT (CREAT), Co^{60} activity buildup prediction includes 1) Co-base valve replacement effect, 2) Co-base valve maintenance effect, and 3) control rod drive mechanism (CRDM) and main coolant pump (MCP) shaft contribution. CREAT predicted that the main contributor of Co activity buildup was the corrosion-induced release of Co from the steam generator (SG) tubings. With new SG's tubed with alloy 690, Korean Next Generation Reactor (APR-1400) is expected to have about 64 % lower Co activity on SG surface. The use of all Co-free valves is expected to cut additional 8 % of activity which is only marginal.

INTRODUCTION

Occupational radiation exposure (ORE) has been decreasing steadily in a few years. Under these trends, more intensive regulations, such as ICRP 60 [1] and 10 CFR 20 [2] are applied. Radioactive nuclei, such as Co^{58} and Co^{60}, deposited on out-of-core surfaces in a PWR primary coolant system by the transport of soluble or particulate crud through the primary coolant, are major sources of ORE to plant maintenance personnel and act as costly impediment to prompt and effective repairs. The valve hard-facing alloys exposed to primary coolant are considered as one of the main Co sources. In KNGR design, the life of plant is aimed to 60 years and resultant accumulated radiation effect will be more severe. Therefore, it is needed to adapt candidate materials and make effort to reduce ORE. To prove the applicability of candidate materials to PWR environment, a performance testing should be conducted in a simulating condition. And in the case of adapting Co-free valves a quantitative evaluation for activity reduction is needed.

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HOT LOOP TEST FOR GATE VALVES

Experimental

As a wear and corrosion resistant alloy, Fe-base and Ni-base superalloys have been developed as Co-free alternatives to Stellite 6 (Co-base). In the developed alloys, it was proposed that NOREM 02 (Fe-base) and Deloro 50 (Ni-base) had the equivalent performance to Stellite 6. The developed alloys, however, were not proven for wear resistance under operating conditions. Gate valves prone to degradation by galling in high stress environment were selected for hot functional tests. The cobalt release rate of a gate valve is less than that of a swing check valve. But their areal fraction in PWR primary loop is the highest among valves and this valve is closely related to safety.

Gate valves are used to open or close a fluid flow. A disc like wedge or gate is used to close flow by moving up and down in normal direction to fluid flow as shown in Fig. 1. NOREM 02 and Deloro 50 were selected as candidate alloys and double-disc 3/4" gate valves were purchased from a nuclear valve manufacturer. Fig. 1 shows the cross sectional view of the used gate valve. Double-disc gate valve has two separate and independent discs and as a pressure difference between two discs increases, the load applied to seat and disc also increases. Therefore, the double-disc gate valve can maintain a low leak rate although the pressure difference increases. Whenever the testing gate valve is closed and the disc contacts the seat, the disc rotates at some degree. It makes disc surface to be uniformly degraded by wear and prevents local wear and resultant leakage problem. Valve actuators used to operate gate valves were motor operated.

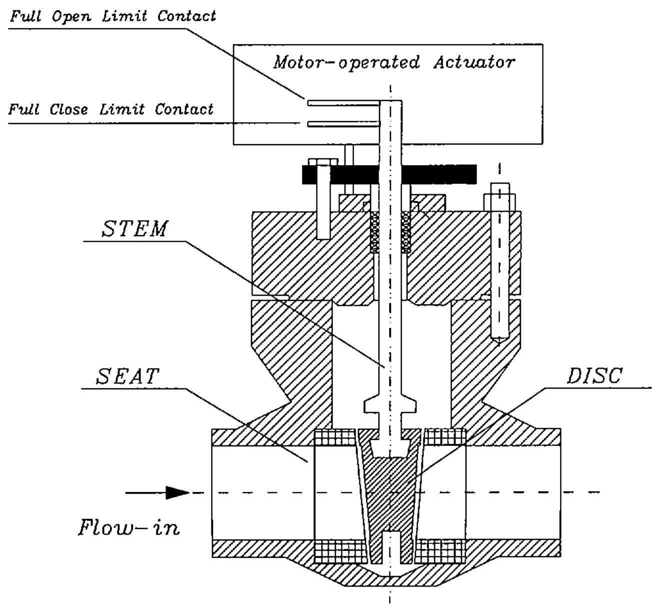

Figure 1. Cross sectional view of the 3/4" test gate valve.

To evaluate the performance of Co-free valves in a PWR primary water condition, a simulation loop, SCOTL has been developed. Fig. 2 shows the schematic diagram of SCOTL. The high temperature/high pressure (HT/HP) water was circulated at a high flow rate of about 40 L/min by a HT/HP centrifugal pump into a 3/4" OD 316 stainless steel (SS) tube. The linear velocity of flowing water was about 2.3 m/s. A 3.8 L autoclave was used as the water heater with the maximum power of 5.0 kW. Test solution composed of 400 ppm boron and 0.8 ppm Li of which high temperature pH is 6.9 deaerated with 5% hydrogen gas (nitrogen bal.) was charged by a diaphragm pressure pump and ejected through a back pressure regulator at about 3 L/hr. The solution tank was made of titanium Gr. 2 to prevent contamination of test solution by tank corrosion. Pressure was maintained at 13.4 MPa with 0.7 MPa fluctuations and the solution temperature was in the range of 277 to 283 °C. The two test valves were connected in parallel between an inlet and outlet header, and each valve was fitted with a motor operated actuator to facilitate stroke cycling of the valves, as shown in Fig. 2.

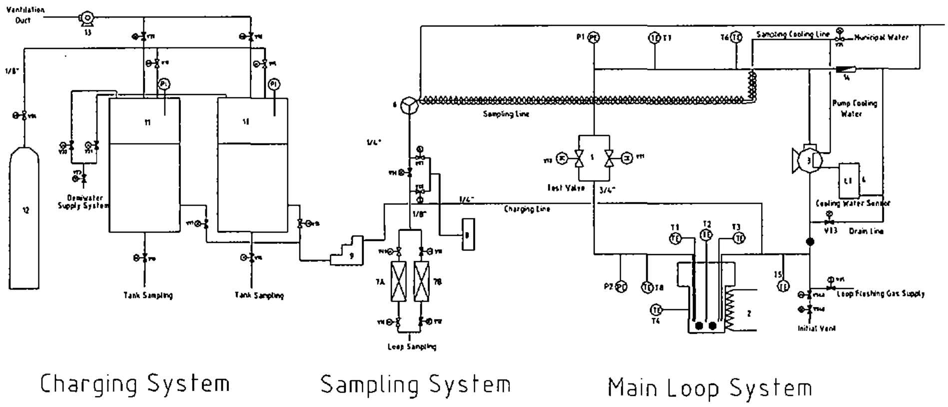

Figure 2. Schematic diagram of HT/HP valve test loop.

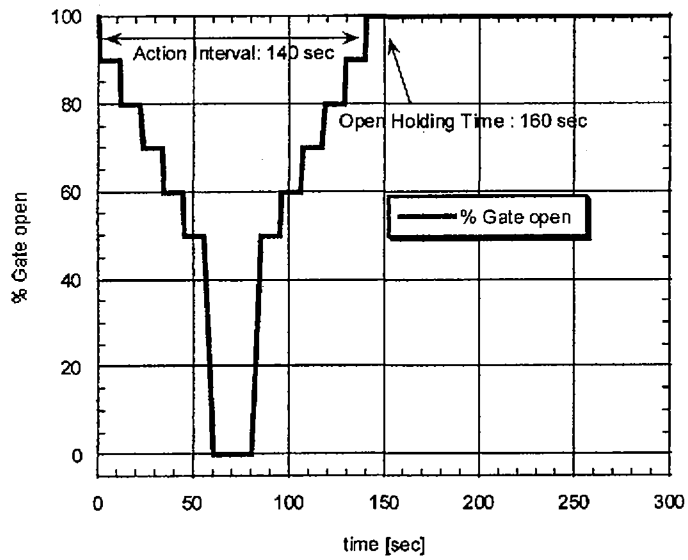

Figure 3. Fraction of gate opening as a function of time during one cycle.

Cycling of each valve was performed in one open-and-close cycle per every 300 seconds as shown in Fig. 3. To prevent abrupt pressure instability of the test loop, the initial valve closing and the final opening were performed step by step. But to simulate a severe wear condition occurred between disc and seat, the valve was continuously operated below the middle position. The other work [3, 4] reported that the rate of cycling was 40-50 cycles for every week of loop operation, and the average period of cycling could be estimated as about 14,000 seconds. The period of cycle in this work was much shorter than that of other work. The passivation current density of 304 SS in 288 °C water as a function of time was measured [5]. It was observed that the corrosion current density after 300 seconds of passivation was about 1,000 times smaller than the beginning bare metal condition [5]. 300 seconds, therefore, was considered enough time to form passive oxide layer on the disc surface.

Prior to cycling test, a cold leak test was performed. The leak rate criterion for nuclear grade valves was 1.125 mL/hr [3]. For the valves satisfying the leak rate criterion, the 1,000 cycles test was conducted. Posterior to 1,000 cycles test, the cold leak test was performed again and the valves were disassembled to examine the surface of valve disc. For the valves passing the leak test, another 1,000 cycles test was conducted. And after 1,000 cycles test, the final cold leak test was performed. A programmed controller governing the valve actuators assured consistent stroke cycling from valve to valve.

Results & Discussions

Fig. 4 shows the cold leak test results prior to cycling test for NOREM 02 and Deloro 50. It was observed that the leak rate of NOREM 02 became smaller as pressure increased and in the range of higher than 1,000 psig NOREM 02 satisfied the criterion. There was no leakage for Deloro 50 in entire pressure range. The pressure dependency of NOREM 02 leak rate could be interpreted that the tightness between disc and seat was improved as the pressure difference between inlet and outlet of valve increased and as a result of that the leakage was reduced. After 1,000 cycles test, the cold leak test was performed for NOREM 02 and Deloro 50. As described in Fig. 5, NOREM 02 showed a heavy leakage even at low pressure, while no leakage was observed in case of Deloro 50. Another 1,000 cycles test was performed for Deloro 50 and Stellite 6. After 1,000 cycles test, the cold leak test was performed and no leakage was observed in the entire range of pressure for two valves.

To examine the surfaces of hardfaced alloys, the valves were disassembled. Fig. 6 shows the SEM micrograph of the worn surface of NOREM 02 after 1,000 cycles test. The scratch was observed on the surface of NOREM 02 disc and even a small crack was detected at disc edge. Fig. 7 and Fig. 8 shows the SEM micrographs of the disc surfaces of NOREM 02 and Deloro 50, respectively. It was observed that many scratches were detected on NOREM 02 surface, while the surface of Deloro 50 was clean. In Fig. 8, the dark area was considered as oxide layer and it was believed that the oxide layer played an important role for the wear resistance of Deloro 50 in PWR primary water condition.

Murphy et al. [3, 4] performed the cycling tests under simulated PWR conditions for one year. The tested hardfacing iron-base alloys were EB 5183, EVERIT 50, NOREM 01 and NOREM 04, and a gate valve with Stellite 6 was used as a control standard.

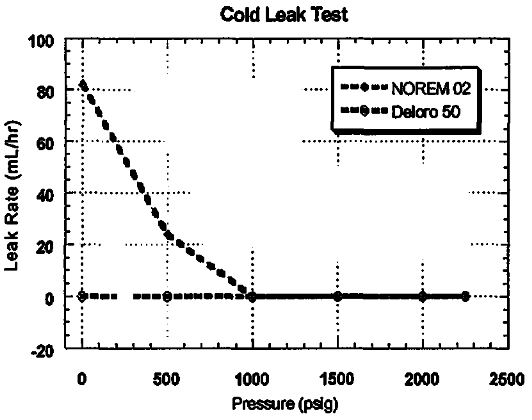

Figure 4. Cold leak test results prior to cycling test as a function of pressure.

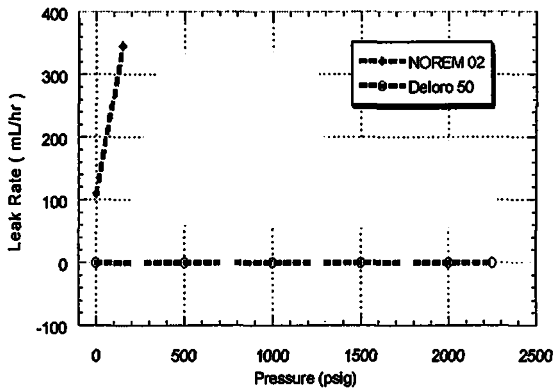

Figure 5. Cold leak test results posterior to 1,000 cycles test as a function of pressure.

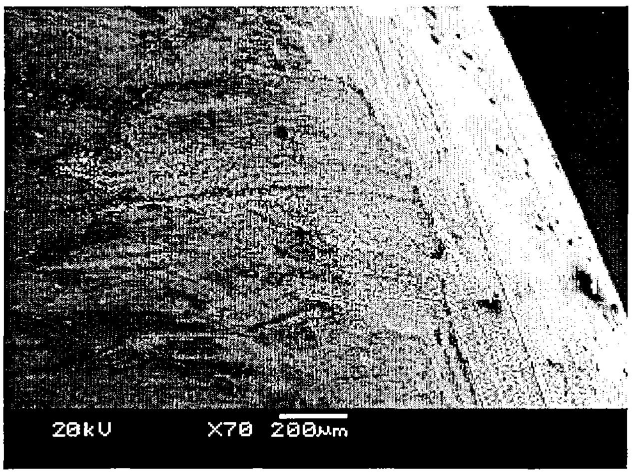

Figure 6. SEM micrograph of the edge of NOREM 02 disc after 1,000 cycles test.

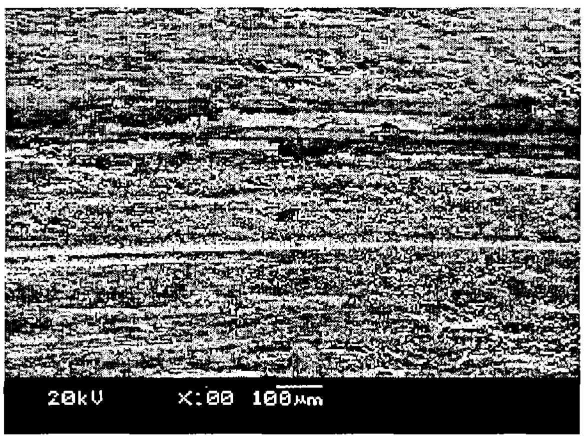

Figure 7. SEM micrograph of the worn surface of NOREM 02 disc after 1,000 cycles test.

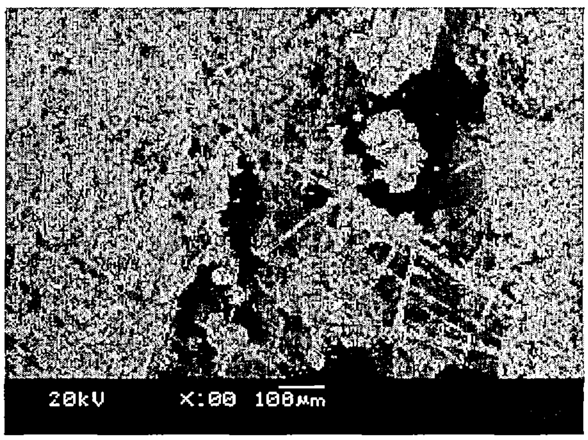

Figure 8. SEM micrograph of the worn surface of Deloro 50 disc after 1,000 cycles test.

During the test period the valves were opened and closed 2,000 times. The performance of the valves was assessed by periodic leak tests and visual and profilometric characterization of sealing surfaces. The various examinations indicated all the iron-base alloys were superior to Stellite 6, but these results are controversial compared with the poor resistance of NOREM 02 to wear observed in this work. Yonezawa et al. [6] performed the endurance test in 300 °C pure water, up to 200 times, using 3" motor gate valves. The leakage rate of NOREM B1 was less than 1 cc/15 min. after 200 times and similar leakage was observed for Stellite 6. They concluded that the performance of NOREM B1 seemed to be the same as that of Stellite 6 in high temperature water. As compared with this work, the number of cycles was not enough to represent the actual wear conditions of gate valves. Kim et al. performed the sliding wear tests for NOREM 02 [7] and Deloro 50 [8] in the temperature range of 25-300 °C. They observed that the wear mode of NOREM 02 changed abruptly to severe adhesive wear at 190 °C and galling occurred above 200 °C and concluded that the development of strain-induced α' martensite plays an important role in the wear resistance of NOREM 02 hardfacing alloy. And they concluded that if the temperature was high enough to meet the oxidative wear condition, Deloro 50 could be used as hardfacing material for nuclear power plants valves even under the high contact stress of 207 MPa. Kim's results supported well the wear behaviors of NOREM 02 and Deloro 50 observed in this hot-loop test.

NUMERICAL MODEL FOR ACTIVITY BUILDUP

Model Development

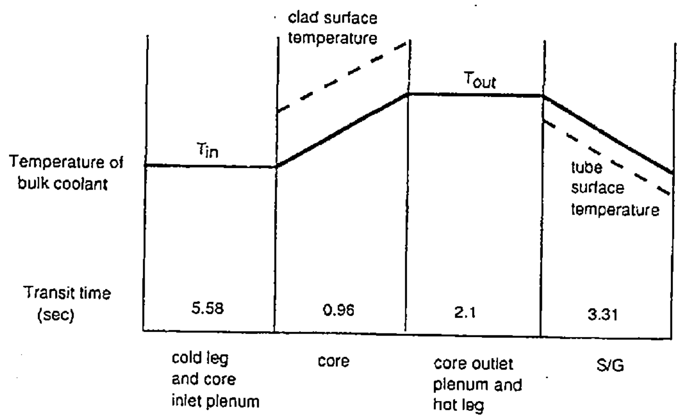

Figure 9. Temperature distribution of primary coolant as a function of location [9].

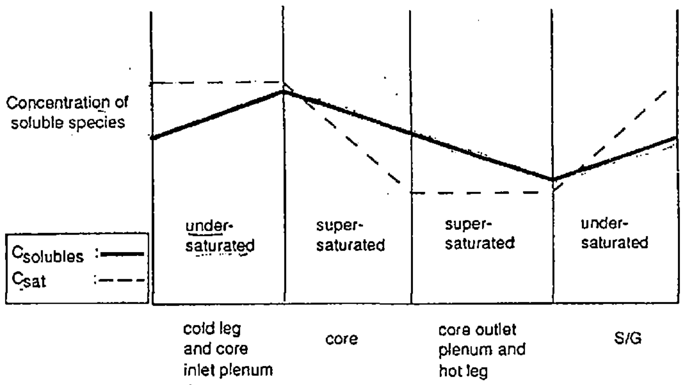

Figure 10. Concentration of soluble crud in primary coolant as a function of location [9].

To predict and estimate the activity buildup in a PWR, three computer codes, such as CORA, PACTOLE, and CRUDSIM-MIT have been developed. CRUDSIM-MIT uses a simple model as compared with CORA and PACTOLE and it only concerns Co^{58} and Co^{60} as radioactive elements [9]. It assumed that the transport of crud and activity were mainly caused by solubility difference. In this work, CRUDSIM-MIT having relatively high accuracy of prediction results and a simple concept was selected as base model.

Fig. 9 shows the PWR primary coolant temperature variation with locations. And Fig. 10 shows the concentration of soluble crud in PWR primary coolant with locations. The local temperature variation of a primary coolant was so fast that the crud concentration in the coolant alternated between super-saturated and under-saturated states. Therefore, it was considered that the crud on SG tube surfaces would be dissolved into the under-saturated coolant and the dissolved crud in the super-saturated coolant would be precipitated on the cladding surfaces. This is the mechanism of soluble crud transport in a PWR primary coolant system. CRUDSIM-MIT also considered the transport of particulate crud. Because CRUDSIM-MIT considered only the corrosion of SG tubings as the source of cobalt input, to evaluate the effect of hardfaced parts with Stellite on activity buildup CRUDSIM-MIT was modified. The corrosion products of SG tubings are dissolved into the coolant as soluble species, but the cobalt from the hardfaced parts can be released by two manners; wear and corrosion. It is assumed that the cobalt is released with particulate form by wear and soluble form by corrosion. The developed model, CREAT introduced some assumptions in crud transport model as below.

1. The cobalt is released from SG tubings, valves, CRDM, and MCP shafts by wear and/or corrosion.
2. The cobalt released by wear has a particulate form and this is deposited on SG and core surfaces according to area ratio. The cobalt released by corrosion is soluble and to represent this effect the corrosion rate of SG tubings is modified.
3. Considering the area fraction, the activity buildup on the surfaces of hardfaced parts can be neglected.
4. Very effective ion-exchange resins are used in PWR primary circuit, but the flow rate through the ion-exchange resins is very low compared with the primary coolant flow rate. And the effect of purification on the crud transport and activity buildup is neglected.

CRUDSIM-MIT assumed that only Fe crud moved through the coolant and Co^{58} and Co^{60} were embedded in the Fe crud as minor elements. The activation coefficient was defined only for the Fe crud. But CREAT defined a new activation coefficient for the Co crud released as particulate form by the wear of hardfaced parts.

Table 1. Annual cobalt release rate of each part with various plant design and conditions.

Classification		Co Release Rate (g/yr)												
		SG	SG corrosion rate, kg-Fe/day	CRDM		Valve maintenance	Valve		MCP shaft		Total		
				Wear	Corrosion		Wear	Corrosion	Wear	Corrosion	SG	Non-SG	Sum
H. Ocken[1]	CE[3]	33	0.04	5	7.52	10	2.3	1.44	0.2	1.71	33	28.17	61.17
	W[4]	42	0.0509	1.80	2.71	10	6.82	4.26	0.37	2.74	42	28.7	70.7
	PWR	33~55	0.04~ 0.0667	0.79~ 1.99	1.20~ 3.01	10	1.5	0.937	0.02	0.18	33~55	14.63~ 17.637	48.86~ 72.64
C. J. Wood[2]	CE[3]	33	0.04	5	7.52	10	2.3	1.44	0.2	1.71	33	28.17	61.17
	W[5]	50	0.006	2.39	3.61	10	3.07	1.93	0.37	1.71	50	23.08	73.08
CREAT	KSNPP	33	0.04	2	3.01	10	2.3	1.44	0.2	1.71	33	20.66	53.66
	KNGR[6] (Alloy 690)	10.532	0.006	2	3.01	10	2.3	1.44	0.2	1.71	10.532	20.66	31.192
	KNGR (good cleaning)	10.532	0.006	2	3.01	2.9	2.3	1.44	0.2	1.71	10.532	13.56	24.092
	KNGR (Co-free valve)	10.532	0.006	2	3.01	0	0	0	0.2	1.71	10.532	6.92	17.452

1. Reference [11].
2. Reference [12].
3. Millstone 2.
4. Beaver Valley 1, Reference [13].
5. Trojan 2.
6. Korean Next Generation Reactor (APR-1400)

In a new code, CREAT, the improved parts are as follows.
1. It is possible to evaluate the contribution of hardfaced parts with Stellite to activity buildup during operation and maintenance.
2. The Co^{59} content in SG tubings can be changed and the effect on activity buildup can be evaluated.
3. The refueling time can be changed.
4. The corrosion rate constant of SG tubings is not fixed, one of input variables.

Results & Discussions

CREAT focused on the prediction of activity variation with cobalt release rate. CREAT predicted results were verified by using 900 MWe French plant data reported by Metge et al. [10]. French plant was very similar to Westinghouse 3-loop model so that the annual cobalt release rate of each part was estimated by using Westinghouse data reported by EPRI. Table 1 shows the annual cobalt release rate for each plant design and condition. For seven plants with different Co^{59} content in SG tubings, Co^{60} activity on the surface of SG tubings and primary pipings was measured at the end of 3rd cycle. Fig. 11 shows the CREAT prediction results in comparison of the measurement results. The prediction results showed a good agreement with the measurement results within maximum 25 % error. Assuming Co content in SG tubings was 470 ppm, the contribution of SG tubings to the total Co^{60} activity was estimated as about 75%. CREAT also predicted the Co^{60} activity variation with time assuming the Co content in SG tubings was 500 ppm. Fig. 12 shows the prediction results in comparison of the measurement results by Metge. CREAT overestimated the Co^{60} activity variation with time up to 50 % as shown in Fig. 12. The activity variation with time can be sensitively affected by corrosion rate or nuclear reaction cross sections. But these parameters were verified in earlier work [9]. Therefore operating parameters, for example water chemistry can be considered. Chemical cleaning during plant shutdown was one of possible operating parameters. The shutdown chemistry is the procedure that during cold shutdown boric acid and hydrogen peroxide are ejected to increase acidity and oxidizing power, which leads to the dissolution and release of the corrosion products from the surface. It seems that the French plant data included the effect of shutdown chemistry. Assumed that 20 % activity reduction was expected by applying shutdown chemistry, CREAT prediction results showed a similar behavior to the plant data as shown in Fig. 12.

Based on the verification results of CREAT, the activity buildup of Korean Next Generation Reactor (APR-1400) was predicted under various conditions. As a reference case, Westinghouse 900 MWe plant, designated as 'Kori #4', was selected. It was assumed that the Co content in alloy 600 SG tubings was 470 ppm and the solution pH was 6.9 and one cycle was composed of 12 months operation and 2 months maintenance. The activity buildup was predicted to the end of 40 years life. In the case of Korean Next Generation Reactor (APR-1400), it was assumed that the Co content in alloy 690 SG tubings was 150 ppm and the corrosion rate of alloy 690 was the same to alloy 600. To consider the recent trend of primary water chemistry control, the prediction was performed for pH of 7.3 as well 6.9. One cycle was assumed as the 18 months operation and 2 months maintenance to consider high burn up of fuel and the activity buildup was predicted to the end of 60 years life. The annual cobalt input of each part was referenced from CE System 80 data designated as 'KSNPP'. In CE System 80 data, the Co release rate by wear of hardfaced parts was reported but not by corrosion. In this work, it was assumed that the ratio of the Co release rate by wear and corrosion was the same as Westinghouse.

Table 1 summarizes the cobalt release rate by wear and corrosion of each part for various plant design and conditions. The Co release rate by valve maintenance was much higher than that by wear and corrosion of valves. It was reported that the post-maintenance cleaning for valves could reduce the cobalt ingress by maintenance effectively. Therefore, the post-maintenance cleaning case was included. The effect of shutdown chemistry was not considered in this prediction because the quantitative evaluation about the shutdown

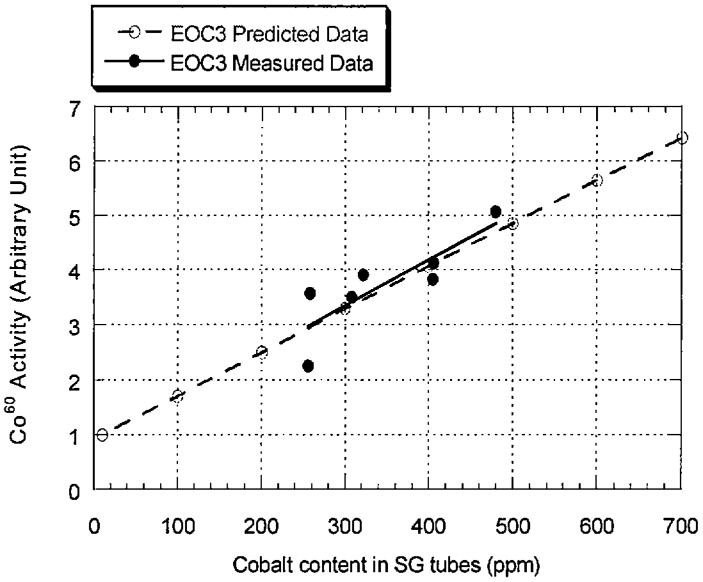

Figure 11. CREAT prediction results in comparison of the French plant data for the surface activity as a function of Co content in SG tubings.

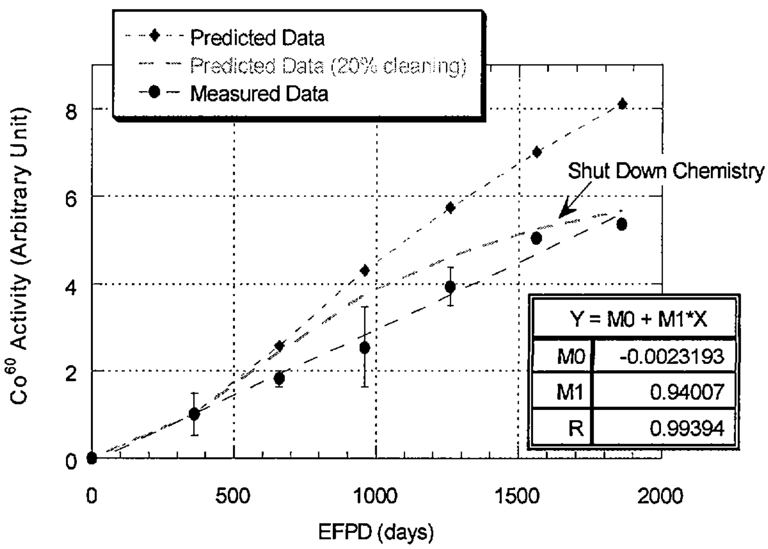

Figure 12. CREAT prediction results in comparison of the plant data for the Co^{60} activity variation with time.

chemistry was not enough up to now. But it will be possible to consider the shutdown chemistry if the evaluation and testing are performed.

Table 2. Co activity prediction results at the end of plant life with various conditions.

		Surface [uCi/cm^2]			Overall [Ci]			Tot. Co input [g/yr]
		Co^{60}	Co^{58}	Total	Co^{60}	Co^{58}	Total	
PH=6.9 @300 °C 1200ppm B + 2.2ppm Li	Kori #4	22.8	8.8	31.6	4360	1680	6040	70.7
	KSNPP	18.2	6.6	24.8	3790	1380	5170	53.7
	KNGR[1] (690 470ppm Co)	26.1	5.3	31.4	7570	1540	9110	53.7
	KNGR (690 300 ppm Co)	18.7	5.3	24	5430	1540	6970	41.7
	KNGR (690 150 ppm Co)	12.2	5.3	17.5	3530	1540	5070	31.2
	KNGR (690 Good cleaning)	11.3	5.3	16.6	3270	1540	4810	24.1
	KNGR (690 Co-free valves)	9.7	5.3	14.7	2820	1540	4360	17.5
PH=7.3 @300 °C 400 ppm B + 2.06 ppm Li	KNGR (690 470 ppm Co)	20.2	2.4	22.4	5800	683	6483	53.7
	KNGR (690 300 ppm Co)	15.8	2.4	18.2	4590	683	5273	41.7
	KNGR (690 150 ppm Co)	12.1	2.4	14.5	3520	683	4203	31.2
	KNGR (690 Good cleaning)	9.1	2.4	11.5	2630	683	3313	24.1
	KNGR (690 Co-free valves)	6.3	2.4	8.7	1820	683	2513	17.5

1) Korean Next Generation Reactor (APR-1400).

Table 2 summarizes the cobalt activity prediction results for KNGR, KSNPP, and reference case (Kori #4). The activity of Co^{58} and Co^{60} becomes comparable usually at the end of 5th or 6th cycle. Fig. 13 shows the activity buildup behavior of Co^{58} and Co^{60} predicted by CREAT and shows a similar behavior as mentioned above. Fig. 14 and 15 show the Co^{60} and Co^{58} activity buildup trends for various plant conditions, respectively. Assumed that the total activity was the sum of Co^{58} and Co^{60}, the total activity buildup trends for various plant conditions were plotted in Fig. 16. The contribution of Co^{58} to activity buildup was about 30~60 % at the end of plant life. The KSNPP showed low activity buildup compared with other cases. It was interpreted that as the average coolant temperature increased, the solubility difference between core and SG surfaces decreased and as a result, crud transport decreased. Fig. 17 and 18 show the surface activity and overall activity at the end of plant life as a function of the annual Co input, respectively. As expected, the activity was proportional to the Co input. In the coolant pH of 6.9, KNGR showed that the total surface activity for the case of alloy 690 with 150 ppm Co was 55 % lower than the case of alloy 690 with 470 ppm Co that was a virtual condition. In the coolant pH of 7.3, the activity showed lower values than that of the coolant pH of 6.9. Besides replacing alloy 690 with alloy 600, the additional activity reduction could be acquired by replacing the Stellite valves with Co-free valves. The replacement of CRDM and MCP shafts was not considered due to the safety problem. In Table 3, an activity reduction factor is defined as the normalized activity with the reference case and summarized for various conditions. If non-replacement of Stellite valves, good cleaning, and the coolant pH of 7.3 were applied, the activity reduction factors for the surface activity and overall activity of KNGR were 0.36 and 0.55. If Co-free valves were applied additionally, the values decreased to 0.28 and 0.42, respectively.

In this work, the relative error for the activity prediction as a function of the Co input was 25 % as shown in Fig. 11. The effect of corrosion rate reduction of alloy 690 compared with alloy 600 was not considered in this prediction, but Beslu et al. [14] reported that the corrosion rate of alloy 690 was the half of alloy 600. It was qualitatively concluded that the corrosion rate reduction of alloy 690 would decrease the activity buildup on SG surface.

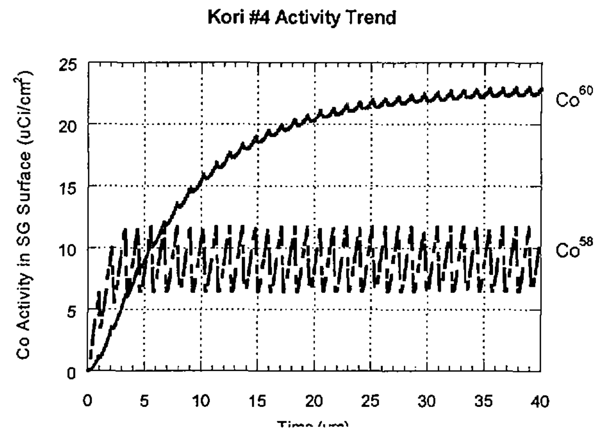

Figure 13. Activity buildup trends of Co^{60} and Co^{58} with time for Kori #4.

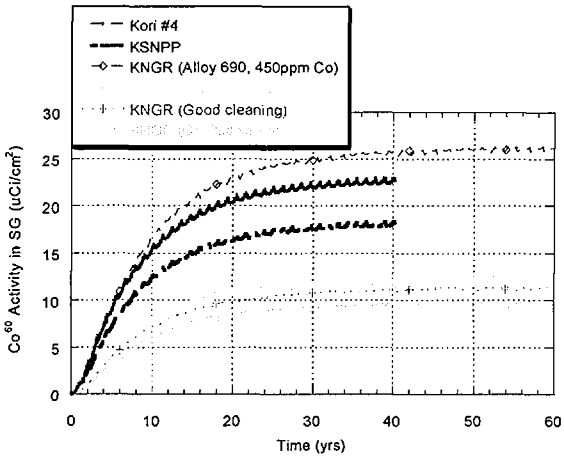

Figure 14. Co^{60} activity buildup trends as a function of time with various conditions.

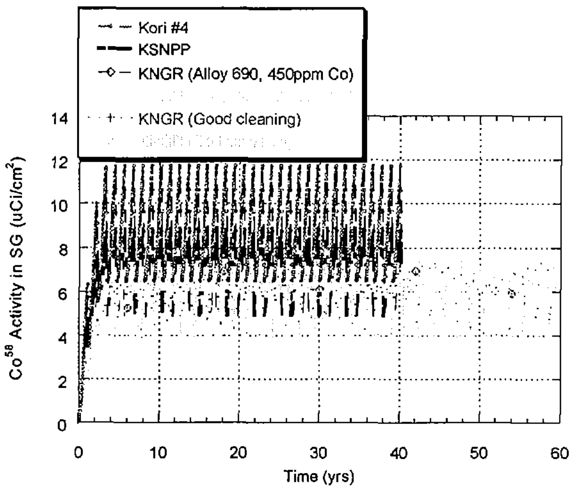

Figure 15. Co^{58} activity buildup trends as a function of time with various conditions.

Table 3. Activity reduction factor for the various conditions.

		Surface Activity		Overall Activity	
		$F^{1)}_{Co^{60}}$	$F_{total\ Co}$	$F_{Co^{60}}$	$F_{total\ Co}$
pH=6.9 @300 °C 1200ppm B + 2.2ppm Li	Kori #4	1.00	1.00	1.00	1.00
	KNGR[2] with Stellite	0.54	0.55	0.81	0.84
	KNGR with Stellite + Good cleaning	0.50	0.53	0.75	0.80
	KNGR without Stellite	0.43	0.47	0.65	0.72
PH=7.3 @300 °C 400 ppm B + 2.06 ppm Li	KNGR with Stellite	0.53	0.46	0.81	0.70
	KNGR with Stellite + Good cleaning	0.40	0.36	0.60	0.55
	KNGR without Stellite	0.28	0.28	0.42	0.42

1) Activity reduction factor (ARF)

$$F = \frac{\text{Total activity in KNGR at the end of 60 yrs life}}{\text{Total activity in Kori \#4 at the end of 40 yrs life and pH of 6.9}}$$

Target ARF=1/2.5=0.4

2) Korean Next Generation Reactor (APR-1400).

CONCLUSIONS

1. To evaluate the performance of Co-free valves in PWR primary coolant, a HT/HP PWR primary coolant simulating test loop, SCOTL has been developed. By using SCOTL cycling test was performed in 280 °C primary coolant for 3/4" gate valves made of NOREM 02, Deloro 50, and Stellite 6.

2. After 1,000 cycles test, NOREM 02 did not satisfy the leak criterion and Deloro 50 had no leakage. Replacing NOREM 02 to Stellite 6, another 1,000 cycles test was conducted. After 1,000 cycles test, no leakage was observed for Deloro 50 and Stellite 6 valves. From the examination of disc surface, it was concluded that NOREM 02 had a poor resistance to wear in a PWR primary coolant condition and Deloro 50 and Stellite 6 showed a good resistance to wear. But to apply the Co-free alloys to PWR valves, more wear experiments under various conditions are needed.

3. To evaluate the effect of Stellite hardfacing materials on the activity buildup, CREAT code has been developed which was based on CRUDSIM-MIT. CREAT prediction results were verified with French plant data and showed a good agreement with the plant data.

4. By using CREAT, the total cobalt activity of Korean Next Generation Reactor (APR-1400) at the end of 60 yrs life was predicted and compared with the reference case, Westinghouse 900 MWe with 40 yrs life. If the SG tubings were replaced to alloy 690 with low Co content, the activity was reduced to 46 % compared with the reference case. And if the Co-free valves were used additionally, the activity was reduced to 28%. Therefore, it appears that the use of alloy 690 and Co-free valves can reduce the activity very effectively.

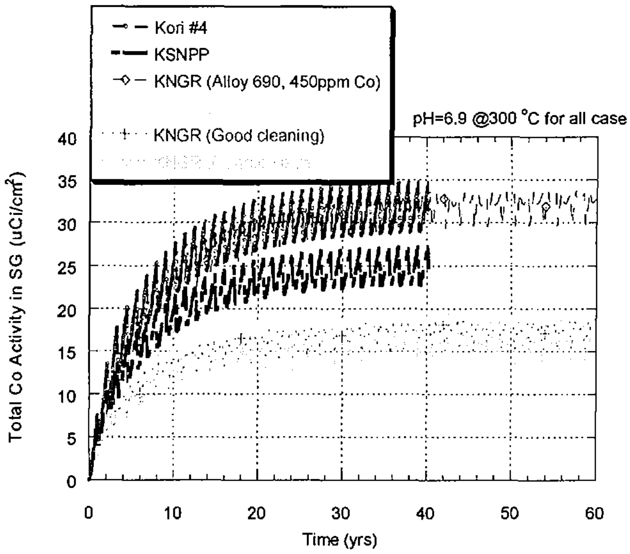

Figure 16. Total Co activity buildup trends as a function of time with various conditions.

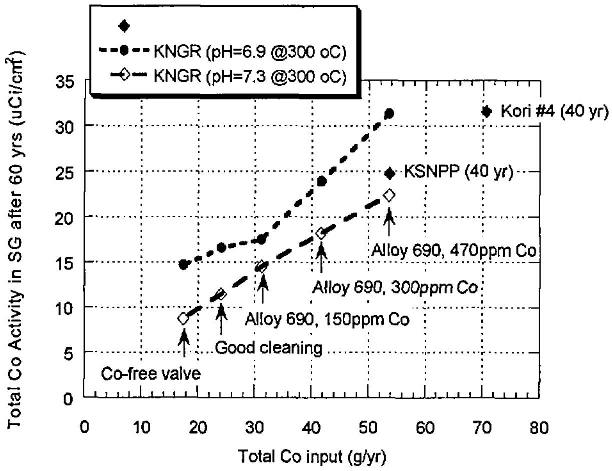

Figure 17. Surface activity at the end of plant life as a function of total Co input.

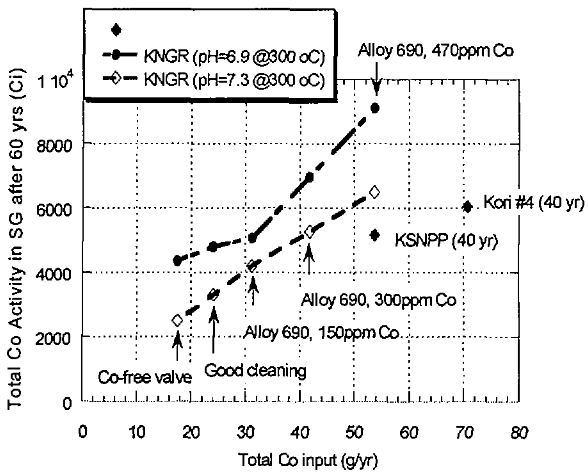

Figure 18. Overall activity at the end of plant life as a function of total Co input.

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