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Analysis of Water Purification Capability of the Spent Fuel Storage Pool Using Consolidated Fuel Storage in Uljin 1&2

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조밀화 핵연료 집합체 저장에 의한 울진 1&2호기의 사용후 핵연료 저장조 정화능력 해석

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Abstract

The radioactivity in the spent fuel storage pool is calculated to ensure to maintain its concentration below the permissible limit, when the storage capacity of Uljin nuclear power plant unit 1&2 is extended from 9/3 to 32/3 core using consolidated fuels in maximum density rack (MDR). For this evalulation, two models to calculate the spent fuel pool activities on the continuous and intermittent operating its purification system are developed and these results compared, The results of above two cases show that the current water purification system can not guarantee the radioactivity concentration below the design limit, 5×10^{-4} μCi/ml, for the extention to 32/3 core. Therfore, it has been concluded that a modification of the current purification system is necessary to extend the spent fuel storage capacity with the above method. The alternative way suggested in this study is to increase the number of cation bed demineralizers.

요 약

울진 1&2호기의 사용후 핵연료 중간저장을 위한 기존 저장조용량확장 방안으로서 maximum density rack (MDR)에 consolidated fuel을 저장하여 현 9/3 노심에서 32/3으로 확장할 경우 방사능 농도가 적정기준 이하로 유지될 수 있는지 여부를 분석하였다. 이를 위해 본 연구에서는 정화계통의 연속적 운전방식과 주기적 운전방식에 대한 저장용수중의 방사능 농도계산을 위한 두가지 계산 모델을 만들어 상호비교 하였다. 이 결과 두 경우 모두 32/3 노심저장에 대하여 기존 정화계통으로는 기준치인 5×10^{-4} μ Ci/ml 이하로 유지시킬 수 없었다. 따라서 기존의 시설변경이 불가피하며 그 방안으로 사용후 핵연료 저장조에서의 양이온 탈염기 수를 증가시키는 방법이 타당한 것으로 나타났다.

1. Introduction

As the reprocessing policy for spent fuel is deferred infinitely and the construction of nuclear power plants is increasing, the spent fuel storage pools will be jampacked around the end of 1980's for the case of Kori 1 and after the middle of 1990's for the other units in Korea. Therefore, a plan for the extension of the current spent fuel storage capacity or the construction of a new interim storage facility must be established before the shortage of at-reactor storage capacity should occur. Especially, its extension is pressing since each storage capacity of Uljin 1&2, is only 9/3 core.

As a part of study on the extension of the spent fuel storage capacity from 9/3 to 32/3 core by storaging the consolidated fuel in the maximum density rack for Uljin 1&2, it has to be evaluated whether, in this case, the current purification system can maintain the storage pool activity below the accept limit or not. For the purpose of this evaluation, two calculational models were developed. One is to calculate spent fuel pool activities on the continuous operating its water purification system and the other on the intermittent operating. The design criteria[1] for the water purification system in the spent fuel storage pool is such that ; during normal operation of the purification system, the annual averaged specific activity in the spent fuel pool should be maintained below 5×10^{-4} μCi/ml. In addition, operation of the purification system should be able to bring down the activity below the basis limit within 72 hours.

2. Description of the calculational model

1) Backgrounds of calculation

It is assumed that upon shutdown for a refueling, the reactor coolant system is cooled down for a period of approximately 100 hours. During this period, the production rate of radioactive nuclides in the reactor coolant is functions of the fraction of defective fuel rods, the escape rate coefficients of nuclides, and the radioactive concentration in the fuel. On the other hand, the primary coolant lets down through the purification filter, purification ion exchanger, gas stripper, and volume control tank. In this calculation, the defective fuel rods are assumed to be uniformly distributed throughout the core ; thus, the fission product escape rate coefficients are based upon Preliminary Safety Analysis Report (PSAR) for Uljin 1&2. The parameters used in the calculation of reactor coolant activities are given in Table 1. The primary leakage-out and gas stripping are excluded.

Table 1. Parameters Used in the Calculation of the Reactor Coolant Activity[3),6),7)]

1. Reactor coolant mass : 1.841E05 Kg
2. Refueling water cavity volume : 1335m^3
3. Chemical volume and control system
   - Purification flow rate
     - ○ during normal operation : 13.6m^3/hr
     - ○ during shutdown : 27.2m^3/hr
   - Filter
     - ○ decontamination factor : Cr-1.4 Mn-4.2 Fe-3.2 Co-5.5
   - Mixed bed demineralizer
     - ○ decontamination factor : noble gas-1; Y, Mo, Cs-1; other isotopes-10
   - Cation bed demineralizer
     - ○ purification flow rate
       during normal operation 1.36m^3/hr
     - ○ decontamination factor : noble gas-1; Y,Mo,Cs-10; other isotopes-1

After reducing somewhat reactor coolant activity, the coolant above the reactor vessel flange is partially drained. The reactor vessel head unbolted and the refueling water cavity are filled with the incoming water from the refueling water storage tank. The remaining reacor coolant volume containing radioactivity is then mixed with water in the refueling cavity. The discharged fuel assemblies from the reactor core are transferred to the spent fuel storage pool through the fuel transfer tube. At this time, part of the refueling cavity water containing radioactivity enters the spent fuel pool through the fuel trasfer tube. After refueling, the spent fuel pool is isolated and the refueling cavity water is returned to the refueling water storage tank.

Because the temperature of the handled and stored spent fuel is much lower than reactor operating temperature, the fuel gap activity escape rate coefficient for the spent fuel located in the spent fuel pool is extremely low.[2] Most of the activity escapes from the defective fuel elements during shutdown and cooldown of the reactor prior to removal of the reactor vessel head. If the significant release from failed fuel are detected, the defective fuel elements will be isolated in a separate container so that the released activity negligibly contributes to the specific activity in the spent fuel pool water. Thus the primary source of radioactivity in the spent fuel pool water, after refueling operations have been completed, is due to displaced activation product (i.e., crud) from the surface of the spent fuel assemblies and radioactive nuclides incoming from the refueling cavity through the fuel transfer tube. The spent fuel pool water purification system consists of the skimmer/strainer to remove the floating matters on the surface of the spent fuel pool as well as the purification components with mixed and cation bed demineralizers and filters. But the skimmer/strainer are excluded in this calculation. The parameters used in the calculation of spent fuel pool activities are given in Table 2.

Table 2. Parameters Used in the Calculation of the Spent Fuel Pool Activity[3),6),7)]

1. Spent fuel pool volume : 1326m^3
2. Purification rate
   ○ during normal operation : 60m^3/hr
3. Filter
   ○ decontamination factor :
      Cr-1.4
      Mn-4.2
      Fe-3.2
      Co-5.5
4. Mixed bed demineralizer
   ○ decontamination factor :
      Noble gas-1
      Y, Mo, Cs-1
      other isotopes-10
5. Cation bed demineralizer
   ○ decontamination factor :
      noble gas-1
      Y, Mo, Cs-10
      other isotopes-1

3. Assumptions used in this study are as follows

1) The fission products escape rate coefficients in the spent fuel pool are 10^{-3} times that in the reactor coolant during normal operation. (See Table 3.)[3), 7)]
2) The fission products and corrosion products activities in the reactor coolant are referred to PSAR for Uljin 1&2.[3)]
3) On normal operation, the fraction of the fuel rods with defective cladding is 1% corresponding to design condition in the reactor coolant and the spent fuel pool.
4) After shutdown, the reactor coolant purification system operates duri 3 days.[4)]
5) It takes 30 minutes to transfer a discharged fuel assembly to the spent fuel storage pool.
6) The life time of the nuclear power plant is 3 years.

Table 3. Escape Coefficients in the Spent Fuel Pool.[3),7)] Unit : Day

Fission products :	
Noble gas	5.616E-06
Halogen and Cs	1.123E-06
Mo	1.728E-07
Te	8.640E-08
Sr, Ba	8.640E-10
La, Y, Zr, Rb	1.382E-10
Nb, Tc, Ru	1.382E-10
Corrosion products :	
Cr-51	6.05E-08
Mn-54	1.40E-07
Fe-55,59	1.97E-07
Co-59	1.36E-06
Co-60	6.86E-08

4. Calculational methods

The fuel design parameters are referred to nuclear design report (NDR) and loading partern of Uljin 1&2 provided by Korea Electric Power Corporation. (See Table 4.) Based upon above the data, the fission product activities in the spent fuel are computed for each batch using ORIGEN2 computer code.[5)] ORIGEN2 code determines the

Table 4. Fuel Loading Pattern

Batch	operating cycle load.-disch.	EDPD day	Enrich. w/o	Burnup MWD/MTU	No. of fuel assembly
1A	1-1	292	1.80	13821	51
1B	1-2	584	1.80	18931	1
2A	1-2	584	2.40	24529	51
2B	1-3	876	2.40	33185	1
3A	1-3	876	3.10	32570	51
3B	1-4	1168	3.10	37928	1
4A-30A, 30B	2-30	876	3.25	33828	51
4B-29B	2-30	1168	3.25	38330	1
31	29-30	584	3.25	22640	52
32	30-30	292	3.25	9890	52

buildup and depletion of nuclides in the nuclear materials during irradiation and decay using the matrix exponential method. The nuclides contained in the ORIGEN2 data bases have been divided into three segments ; 130 actinides, 850 fission products, and 720 activation products (a total of 1700 nuclides). Therefore, ORIGEN2 code is suitable for calculating the PWR spent fuel composition. After reactor shutdown for a refueling, the reactor coolant should be purified for a while. At this time, the rate of changes in the activity of nuclide i may be written as follow ;

$$\frac{dC_i}{dt}=\frac{D\,\alpha_i A_i}{V}e^{-(\lambda_i+\alpha_i)t}-\beta_i C_i(t) \qquad (1)$$

where

- A_i : radioactivity of nuclide i calculated in the core fuel using ORIGEN2 code
- D : fraction of fuel rods with defective cladding
- α_i : escape rate coefficient of nuclide i
- λ_i : decay constant of nuclide i
- f : CVCS purification flow rate
- DF_i : decontamination factor of nuclide i

$$\beta_i=\lambda_i+f_i=\lambda_i+f(1-\frac{1}{DF_i})$$

- Ci : radioactivity of nuclide i in the reactor coolant
- V : reactor coolant mass
- t : operation time of purification system.

If $C_i(0)=M_i$ at t=0, the solution of eq.(1) can be expressed as follow ;

$$C(t)=M_i e^{-\beta_i t}+\frac{D\,\alpha_i A_i}{V(f_i-\alpha_i)}\left[e^{(f_i-\alpha_i)t}-1\right]e^{-\beta_i t} \qquad (2)$$

Now the spent fuel pool activities are calculated for two following cases.

i) Continuous operation of the purification system

If the spent fuel assemblies for each batch are placed in the spent fuel pool and the radionuclides are continuously released from the spent fuel, the rate of change in activity of nuclide i may be written as follow;

–at n-th year,

$$\frac{dC_i}{dt}=\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]e^{-(\lambda_i+\alpha_i)t}-\beta_iC_i(t) \quad (3)$$

where,

$B_{ik}=D\alpha_iA_{ik}/V$

A_{ik} : the activity of nuclide i in the spent fuel after being discharged at the k-th year

γ : refueling interval of the fuel.

Since $C_i(0)=C_{i,n-1}(\gamma)+M_i$ at $t=0$, the solution of eq.(3) is expressed in the form;

$$C_{i,n}(t)=\left[C_{i,n-1}(\gamma)+M_i\right]e^{-\beta_it}+\frac{\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}}{f_i-\alpha_i}\times\left[e^{(f_i-\alpha_i)t}-1\right]e^{-\beta_it} \quad (4)$$

where $C_{i,n-1}(\gamma)$ in the radioactivity concentration of nuclide i at the end of the $(n-1)$-th year.

ii) Intermittent operation of the purification system

We assume that the purification system has repeated operation/stop at the interval of $\tau_0=\tau/2M$. Then, the equations of the changes in the activity of nuclide i for each interval are given as follow; –at n-th years

● first operation condition

$$\frac{dC_i}{dt}=\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]e^{(\lambda_i+\alpha_i)t}-\beta_iC_i(t) \quad (5)$$

Since $C_i(0)=C_{i,n-1}(\gamma)+M_i$ at $t=0$, the solution of eq.(5) is expressed in the form;

$$C_{i,n,1}(t)=\left[C_{i,n-1}(\gamma)+M_i\right]e^{-\beta_it}+\frac{\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}}{f_i-\alpha_i}\left[e^{(f_i-\alpha_i)t}-1\right]e^{\beta_it} \quad (6)$$

● first stop condition

$$\frac{dC_i}{dt}=\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]e^{(\lambda_i+\alpha_i)\gamma_0}e^{-(\lambda_i+\alpha_i)t}-\lambda_iC_i(t) \quad (7)$$

Since $C_i(0)=C_{i,n,1}(\gamma_0)$ at $t=0$, the solution of eq.(7) is given as follow;

$$C_{i,n,2}(t)=C_{i,n,1}(\gamma_0)e^{-\lambda_it}+\frac{1}{\alpha_i}\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]\times\left[e^{-\lambda_it}-e^{-(\lambda_i+\alpha_i)t}\right]e^{-(\lambda_i+\alpha_i)\gamma_0} \quad (8)$$

● m-th operation condition

$$\frac{dC_i}{dt}=\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)^{\gamma}}\right]e^{-2(m-1)(\lambda_i+\alpha_i)\gamma_0}\times e^{-(\lambda_i+\alpha_i)t}-\beta_iC_i(t) \quad (9)$$

Since $C_i(0)=C_{i,n,2(m-1)}(\gamma_0)e$ at $t=0$, the solution of eq. (9) is given as follow;

$$C_{i,n,2m-1}(t)=C_{i,n,2(m-1)}(\gamma_0)e^{-\beta_it}+\frac{\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}}{f_i-\alpha_i}\times\left[e^{(f_i-\alpha_i)t}-1\right]e^{-2(m-1)(\lambda_i+\alpha_i)\gamma_0}e^{-\beta_it} \quad (10)$$

In addition, we can obtain the activity at the end of the m-th operation condition;

Table 7. The Description of the Intermittent operation Cases

Case	Case No.	No. of cation bed demi.	Operation period
Continuous operation	1	1	–
	2	2	–
	3	3	–
Intermittent operation	4	1	2 months
	5	2	2 months
	6	3	2 months
	7	3	3 months
	8	3	20 days

Table 5. The Fission Product Activities in the Spent Fuel Assemblies for Each Batch Unit : Ci

BATCH	1A	1B	2A	2B	3A	3B	4A–30A	4B–29B	31	32
RB 89	4.141E05	3.771E05	3.769E05	3.378E05	3.866E05	3.460E05	3.860E05	3.513E05	4.498E05	5.617E05
SR 89	4.427E05	4.019E05	4.050E05	5.373E05	4.087E05	3.708E05	4.105E05	3.770E05	4.820E05	5.736E05
Y 91	5.725E05	5.302E05	5.339E05	4.805E05	5.375E05	4.954E05	5.395E05	5.023E05	6.187E05	7.027E05
SR 92	5.929E05	5.554E05	5.499E05	5.138E05	5.630E05	5.169E05	5.610E05	5.216E05	6.235E05	7.412E05
Y 92	5.959E05	5.585E05	5.528E05	5.169E05	5.660E05	5.199E05	5.639E05	5.244E05	6.262E05	7.436E05
Y 93	6.965E05	6.626E05	6.528E05	6.235E05	6.675E05	6.215E05	6.644E05	6.249E05	7.186E05	8.283E05
Y 94	7.061E05	6.779E05	6.663E05	6.442E05	6.811E05	6.391E05	6.775E05	6.415E05	7.211E05	8.148E05
ZR 95	7.870E05	7.724E05	7.697E05	7.434E05	7.748E05	7.455E05	7.753E05	7.482E05	8.173E05	8.309E05
NB 95	7.806E05	7.777E05	7.840E05	7.504E05	7.825E05	7.596E05	7.842E05	7.634E05	8.301E05	7.795E05
NB 97	8.010E05	7.914E05	7.716E05	7.760E05	7.886E05	7.585E05	7.829E05	7.570E05	7.904E05	8.376E05
NB 97M	7.524E05	7.431E05	7.247E05	7.285E05	7.408E05	7.123E05	7.355E05	7.110E05	7.431E05	7.886E05
MO 99	8.671E05	8.658E05	8.435E05	8.602E05	8.623E05	8.376E05	8.559E05	8.345E05	8.471E05	8.770E05
TC 99M	7.591E05	7.580E05	7.385E05	7.531E05	7.549E05	7.334E05	7.493E05	7.306E05	7.416E05	7.686E05
RU103	7.526E05	7.960E05	7.714E05	8.323E05	7.810E05	7.979E05	7.770E05	7.884E05	6.995E05	5.939E05
RU105	5.364E05	5.857E05	5.568E05	6.318E05	5.663E05	5.908E05	5.588E05	5.793E05	4.599E05	3.432E05
TE131	4.299E05	4.341E05	4.199E05	4.342E05	4.277E05	4.192E05	4.239E05	4.165E05	4.101E05	4.092E05
I131	4.868E05	4.908E05	4.754E05	4.923E05	4.841E05	4.749E05	4.802E05	4.717E05	4.630E05	4.562E05
TE132	6.875E05	6.942E05	6.692E05	6.919E05	6.844E05	6.667E05	6.778E05	6.627E05	6.586E05	6.612E05
I132	6.994E05	7.069E05	6.811E05	7.052E05	6.965E05	6.790E05	6.896E05	6.748E05	6.687E05	6.694E05
I133	9.728E05	9.701E05	9.439E05	9.601E05	9.635E05	9.341E05	9.561E05	9.306E05	9.493E05	9.842E05
XE133	9.708E05	9.441E05	9.432E05	9.603E05	9.439E05	9.347E05	9.562E05	9.313E05	9.485E05	9.361E05
TE134	7.771E05	7.581E05	7.476E05	7.668E05	7.668E05	7.309E05	7.628E05	7.318E05	7.907E05	8.774E05
I134	1.060E05	1.056E05	1.030E05	1.053E05	1.053E05	1.019E05	1.045E05	1.016E05	1.042E05	1.089E05
I135	9.075E05	9.074E05	8.835E05	9.018E05	9.026E05	8.770E05	8.957E05	8.735E05	8.855E05	9.147E05
XE138	7.864E05	7.736E05	7.587E05	7.582E05	7.764E05	7.447E05	7.716E05	7.442E05	7.869E05	8.496E05
CS138	8.763E05	8.647E05	8.463E05	8.492E05	8.655E05	8.332E05	8.598E05	8.310E05	8.715E05	9.314E05
BA139	8.585E05	8.491E05	8.301E05	8.355E05	8.486E05	8.176E05	8.429E05	8.160E05	8.504E05	9.032E05
BA140	8.290E05	8.212E05	7.998E05	8.078E05	8.220E05	7.871E05	8.173E05	7.855E05	8.205E05	8.723E05
LA140	8.435E05	8.516E05	8.218E05	8.425E05	8.582E05	8.203E05	8.475E05	8.184E05	8.383E05	9.294E05
CE141	7.908E05	7.801E05	7.681E05	7.649E05	7.796E05	7.551E05	7.785E05	7.547E05	7.933E05	8.282E05
CE143	7.202E05	7.026E05	6.894E05	6.815E05	7.056E05	6.717E05	7.014E05	6.723E05	7.258E05	7.953E05
CE144	3.489E05	4.643E05	5.661E05	5.661E05	5.864E05	5.955E05	5.940E05	6.002E05	5.382E05	3.489E05
PR144	4.051E05	4.705E05	5.341E05	5.721E05	5.920E05	6.010E05	5.995E05	6.056E05	5.432E05	3.537E05
TOTAL	8.173E07	8.193E07	8.040E05	8.227E07	8.252E07	8.048E07	8.204E07	8.022E07	8.093E07	8.310E07

Table 6. The Spent Fuel Pool Activities for Case 1 after 30-th refueling unit : μCi/ml

Table	.0	2.0	15.0	30.0	45.0	51.0	91.0	183.0	365.0
SR 89	.5109E-08	.4283E-06	.4175E-06	.3404E-06	.2776E-06	.2559E-06	.1485E-06	.4250E-07	.3576E-08
SR 90	.2452E-06	.2760E-06	.2808E-06	.2806E-06	.2803E-06	.2802E-06	.2794E-06	.2777E-06	.2742E-06
Y 90	.1676E-08	.1805E-08	.6284E-10	.1291E-11	.2653E-13	.5609E-14	.1777E-18	.7966E-29	.2688E-49
Y 91	.1653E-06	.5215E-07	.3876E-07	.2893E-07	.2159E-07	.1921E-07	.8805E-08	.1464E-08	.4210E-10
ZR 95	.2058E-06	.1145E-06	.1158E-06	.9860E-07	.8398E-07	.7876E-07	.5133E-07	.1918E-07	.2736E-08
NB 95	.1170E-08	.1102E-06	.9915E-07	.7367E-07	.5474E-07	.4861E-07	.2202E-07	.3562E-08	.9697E-10
MO 99	.6411E-04	.5468E-04	.2189E-05	.5304E-07	.1285E-08	.2903E-09	.1428E-13	.1761E-23	.4400E-43
RU103	.1141E-08	.9600E-07	.8918E-07	.6880E-07	.5307E-07	.4784E-07	.2395E-07	.4876E-08	.2092E-09
RU106	.1565E-07	.4439E-07	.4795E-07	.4660E-07	.4529E-07	.4478E-07	.4150E-07	.3485E-07	.2466E-07
I131	.4142E-05	.4575E-03	.1727E-03	.4712E-04	.1285E-08	.7645E-05	.2393E-06	.8295E-10	.1185E-16
TE132	.2687E-06	.3911E-04	.2856E-05	.1170E-06	.4793E-08	.1335E-08	.2663E-12	.8222E-21	.1200E-37
CS134	.5894E-04	.9582E-04	.9551E-04	.9428E-04	.9306E-04	.9258E-04	.8943E-04	.8260E-04	.7058E-04
CS136	.3527E-05	.1405E-04	.7141E-05	.3210E-05	.1443E-05	.1048E-05	.1243E-06	.9223E-09	.5647E-13
CS137	.2617E-03	.2642E-03	.2640E-03	.2637E-03	.2635E-03	.2634E-03	.2627E-03	.2611E-03	.2581E-03
BA140	.1304E-04	.2332E-05	.3884E-06	.1723E-06	.7641E-07	.5519E-07	.6315E-08	.4313E-10	.2242E-14
CE141	.2267E-05	.41793-06	.9718E-07	.7060E-07	.5129E-07	.4514E-07	.1925E-07	.2713E-08	.5622E-10
CE143	.2499E-06	.5171E-07	.6450E-10	.3359E-13	.1750E-16	.8505E-18	.1494E-26	.1089E-46	.1585E-86
CE144	.1292E-05	.2725E-06	.1019E-06	.9830E-07	.9478E-07	.9341E-07	.8475E-07	.6778E-07	.4355E-07
ND147	.3757E-09	.3992E-07	.2064E-07	.8096E-08	.3175E-08	.2183E-08	.1799E-09	.5780E-12	.6757E-17
CR 51	.8541E-06	.1495E-05	.1646E-05	.1140E-05	.7845E-06	.6756E-06	.2495E-06	.2525E-07	.2717E-09
MN 54	.1192E-07	.5335E-07	.6116E-07	.5893E-07	.5678E-07	.5594E-07	.5065E-07	.4031E-07	.2566E-07
FE 59	.2344E-09	.2485E-07	.2618E-07	.2078E-07	.1649E-07	.1504E-07	.8120E-08	.1969E-08	.1193E-09
CO 58	.3340E-08	.1425E-06	.1504E-06	.1299E-06	.1122E-06	.1058E-06	.7155E-07	.2912E-07	.4916E-08
CO 60	.3747E-06	.5055E-06	.5297E-06	.5268E-06	.5239E-06	.5228E-06	.5153E-06	.4984E-06	.4665E-06
TOTAL	.4117E03	.1169E02	.5485E03	.4117E03	.3734E03	.3670E03	.3541E03	.3448E03	.3295E03

In addition, we can obtain the activity at the end of the m-th operation condition ;

$$C_{i,n,2m-1}(\gamma_0)=C_{i,n,1}(\gamma_0)e^{-(m-1)(\lambda_i+\beta_i)\gamma_0}$$

$$\times\frac{1}{\alpha_i}\left[\sum_{j=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]$$

$$\times\left[1-e^{-\alpha_i\gamma_0}\right]$$

$$\times\left[\sum_{j=1}^{m-1} e^{-(2j-1)(\lambda_i+\alpha_i)\gamma_0}e^{-(m-j)(\lambda_i+\beta_i)\gamma_0}\right]$$

$$+\sum_{k=1}^{n}\frac{B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}}{f_i-\alpha_i}$$

$$\times\left[e^{(f_i-\alpha_i)t}-1\right]e^{(f_i-\alpha_i)\gamma_0-1]} \quad (11)$$

$$\times\left[\sum_{j=1}^{m-1} e^{-2j(\lambda_i+\alpha_i)\gamma_0}e^{-(m-1-j)(\lambda_i+\beta_i)\gamma_0}\right]e^{-\beta_i\gamma_0}$$

● m-th stop condition

$$\frac{dC_i}{dt}=\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]$$

$$\times e^{-(2m-1)(\lambda_i+\alpha_i)\gamma_0}e^{-(\lambda_i+\alpha_i)t}-\lambda_iC_i(t) \quad (12)$$

Since $C_i(0)=C_{i,n,2m-1}(\gamma_0)$ at $t=0$, the solution of eq.(12) is given as follow ;

$$C_{i,n,2m}(t)=C_{i,n,2m-1}(\gamma_0)e^{-\lambda_i t}+\frac{1}{\alpha_i}$$

$$\times\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]$$

$$\times e^{-(2m-1)(\lambda_i+\alpha_i)\gamma_0}\left[1-e^{-\alpha_i t}\right]e^{-\lambda_i t} \quad (13)$$

In addition, we can obtain the activity at the end of m-th stop condition ;

$$C_{i,n,2m}(\gamma_0)=C_{i,n,1}(\gamma_0)e^{-m(\lambda_i+\beta_i)\gamma_0}e^{\beta_i\gamma_0}$$

$$\times\frac{1}{\alpha_i}\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]$$

$$\times\left[e^{-\lambda_i\gamma_0}-e^{-(\lambda_i+\alpha_i)\gamma_0}\right]$$

$$\times\left[\sum_{j=1}^{m} e^{-(2j-1)(\lambda_i+\alpha_i)\gamma_0}e^{-(m-j)(\lambda_i+\beta_i)\gamma_0}\right] \quad (14)$$

$$+\frac{\left[\sum_{k=1}^{n} B_{ik}e^{-(n-k)(\lambda_i+\alpha_i)\gamma}\right]}{f_i-\alpha_i}$$

$$\left[e^{(f_i-\alpha_i)\gamma_0}-1\right]$$

$$\times\left[\sum_{j=1}^{m-1} e^{-2j(\lambda_i+\alpha_i)\gamma_0}e^{-(m-j)(\lambda_i+\beta_i)\gamma_0}\right]$$

5. Results

The spent fuel activities are computed for each batch using ORIGEN2 computer code and the results are represented in Table 5. Just after dischare, the activities of spent fuel assemblies for each batch are slightly over 8.0×10^7 Ci. There is little difference between the activities for each batch since the activity is saturated with burnup being over 1000 MWD/MTU. The structure materials except the cladding material (i.e., Zircaloy) are excluded since the spent fuel assembly is disassembled and the fuel rods are only stored in the maximum density racks.

1) Case of continuous operation

Figure 1 shows the spent fuel pool activities of n-th year under continuous operation after n-th refueling. Each case was specified by the number of cation bed demineralizer, 1,2 and 3 respectively. (See Table 7.) For the case when the total of 1668 (32/3 core) assembly places are completely occupied by the spent fuel assemblies over 30 years and the purification system continues its operation, the change in the activity of the storage pool is calculated and shown in Figure. 2. The radioactivities for each nuclides are given in Table 6. As shown in Figures 1 and 2, the storage capacity of spent fuel pool with the current purification system can not be extended to 25/3 core or more. However, Increasing the number of the

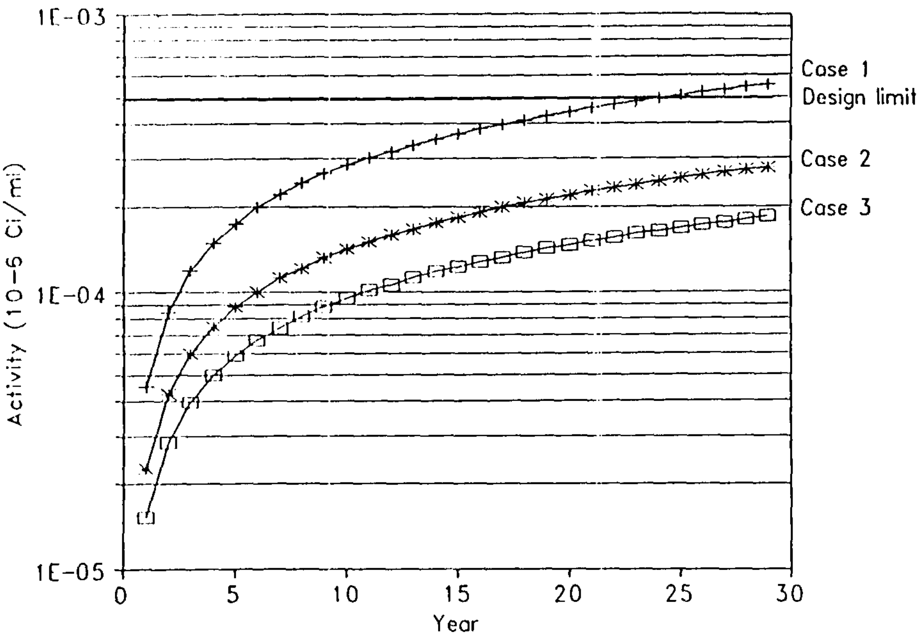

Figure 1. Annual Spent Fuel Pool Activities for Continuous Operation

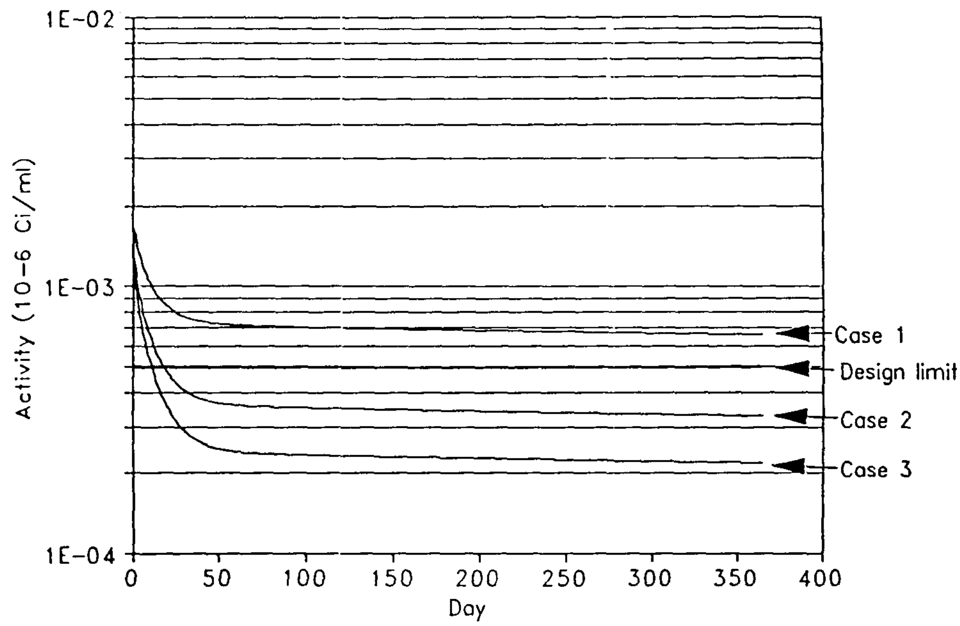

Figure 2. Spent Fuel Pool Activities after 30-th Refueling for Continuous Operation

cation bed demineralizer to 2 or more in the spent fuel storage pool forces its radioactivities to maintain below 5×10^{-4} μCi/ml even though the storage capacity would be extended to 32/3 core.

2) Case of intermittent operation

According to the number of cation bed demineralizer and operation period of the purification system of spent fuel storage pool, each case (Case 4 through 8) was specified and shown in Table 7. Figure 3 shows the spent fuel pool radioactivities of n-th year under intermittent operation ofter n-th refueling. Figure 4 describes the change in the radioactivity of the storage pool after 30-th refueling. As shown in Figure 3 and 4, all cases can not satisfy the design limit. At the start of the purification, storage pool activity is higher than that for a continuous operation. This is due to the radionuclides which have been more accumulated before refueling. The half lives of the major radioactivity sources (i.e., Cs-134 and Cs-137) are longer than those of the other sources so that there is little effect on reducing radioactivity due to a decay during the purification system stop. Instead the radioactivity is on the increase due to the leakage from the spent fuel assembly through defective fuel rods.

6. Conclusion

According to the results for above all cases, the major radioactivity sources in the storage pool are

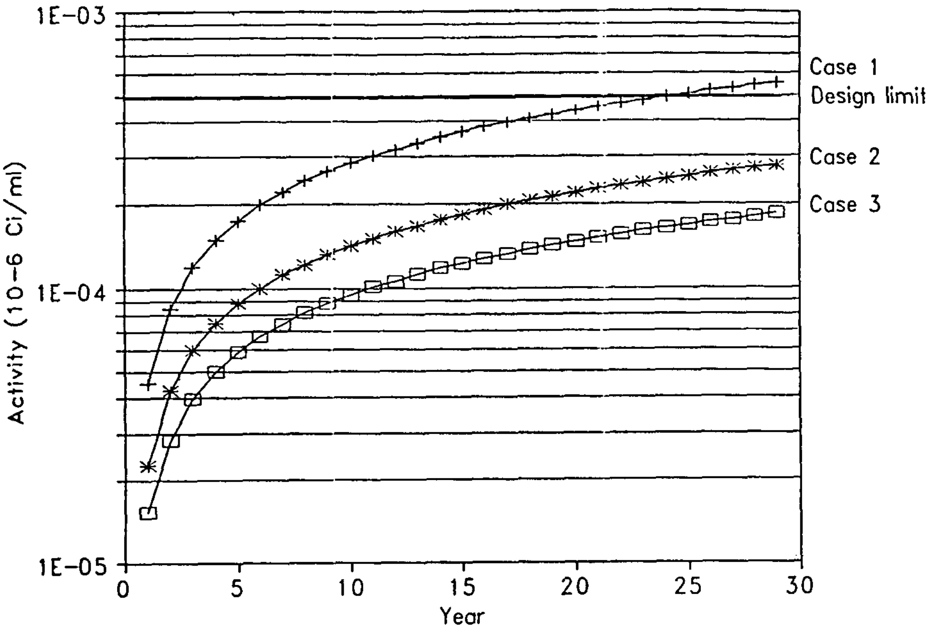

Figure 3. Annual Spent Fuel Pool Activities for Intermittent Operation

fission products such as Cs, I, Sr, Ce, and Y, and corrosion products such as Co, Cr, Fe, and Mn, Among them, Cs-134 and Cs-137 are present for a long time as the major source. The radioactivities for all cases except Case 2 and 3 can not be individually maintained below 5×10^{-4} μCi/ml corresponding to the permissible limit. According to Figure 4, to reduce the operation period makes the maximum radioactivity decrease. From Figure 2, it is shown that, after operation of the purification system, the activity is strictly decreasing until 30 days and slowly thereafter. However, the operation of the purification system can not be able to reduce the storage pool activity below permissible limit within 72 hours for above any cases. Finally, we may conclude that the storage pool activity can not be controled below the permissible limit using the current purification system and intermittent operation for extention of the spent fuel storage to 32/3 core. Although its purification system can not reduce below the limit, 5×10^{-4} μCi/ml within 72 hours, the way which increases the number of cation bed demineralizer was suggested to reduce the radioactivities of spent fuel pool below the limit within 3 weeks. The more the number of cation bed demineralizer, the shorter the time required for spent fuel pool radioactivity to become below the permissible limit.

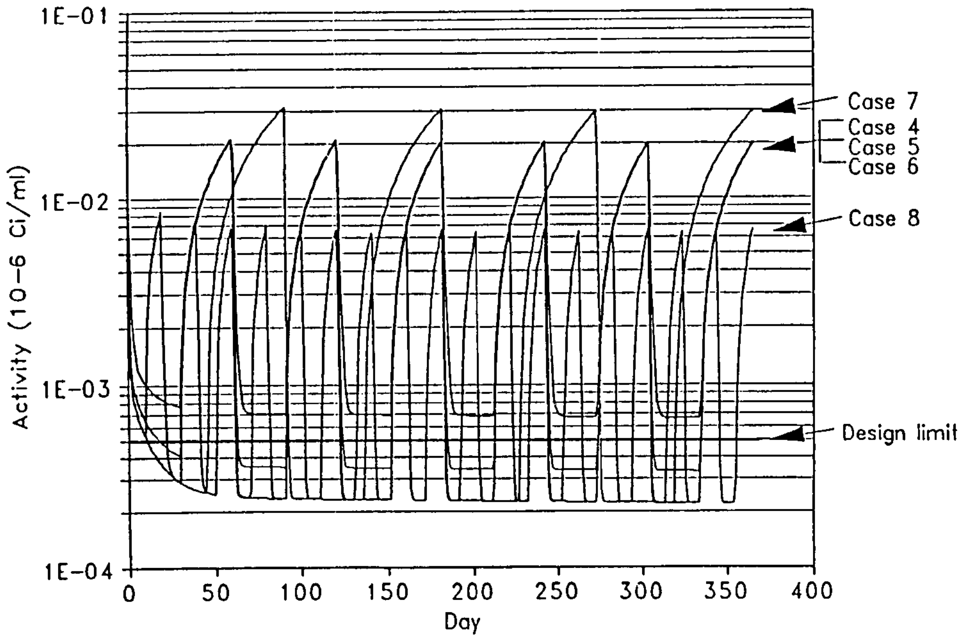

Figure 4. Spent Fuel Pool Activities after 30 -th Refueling for Intermittent operation

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