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Fuel Safety Analysis of the Stagnation Feeder Break Accident for Wolsong1 NPP Loaded with CANFLEX-NU Fuel

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ABSTRACT

The fuel safety analysis of the stagnation break is done for Wolsong1 Nuclear Power Plant(NPP) loaded with CANFLEX-NU fuel. The initiating event, a break in an inlet feeder, can lead to a reduction in coolant flow in the adjacent fuel channel with the channel remaining at power. Depending on the size of the break, a complete stagnation of channel flow can occur, resulting in fuel and channel heatup and channel failure. If the break is somewhat smaller, such that the channel remains intact with a very low flow, it can still lead to significant fuel failure and radionuclide release. Results are presented only for the former event, identified as "stagnation breaks". Feeder breaks can be postulated to occur in any channel, with a wide range of consequences depending on break size, location, and the channel involved. Of this spectrum of events, a break in a high power channel is expected to lead to the limiting release and dose consequences. Thus, to ensure that the worst consequences are covered, a bounding channel is defined as a 7.3 MW(th) channel with a peaked flux shape. To analyze this bounding channel for radionuclide release from fuel, the geometry and parameters for channel O6 are used with power and radionuclide inventories scaled to the defined bounding channel (O6_mod channel) parameters. For a feeder stagnation break with ECC injection available, the total channel fission product release is calculated to be 67,285 TBq or approximately 25% of the total product inventory. This release occurs quickly, in the first 10 - 20 seconds, and prior to channel failure.

1. Introduction

The overall system behaviour following a break in a feeder is very similar to that resulting from a small break in the associated reactor inlet or outlet header. However, the behaviour in the individual affected channel can be quite different. For a particular range of inlet feeder break sizes, the flow in the downstream channel can be reduced sufficiently with no coincident reduction in channel power, that the fuel and channel can heat up rapidly and possibly fail. This heatup is confined to the affected (broken) channel, while the remainder of the core remains adequately cooled as for other small breaks in the primary circuit.

For a very small break in an inlet feeder, the flow in the downstream channel remains in the forward direction and can provide adequate fuel cooling. On the other hand, a complete severance of an inlet feeder causes the downstream channel flow to quickly reverse and thus both the flow from the reactor inlet header and from the downstream channel exit at the break. Channel flow in this case, although in the reverse direction compared to normal flow, still provides adequate fuel cooling. Between these two break sizes, there exists a spectrum of breaks in which progressively lower flow is attained. Over a narrow range of break sizes, the flow in the downstream channel can be more or less stagnated due to a balance between the pressure at the break on the upstream side, and the reverse driving pressure between the break and the downstream end. In the extreme, this can lead to rapid fuel heatup and fuel damage and failure of the fuel channel, similar to that associated with a severe channel flow blockage. Unlike flow blockages there are two pathway for radionuclide releases, the feeder break and the failed channel. The former pathway leads directly to containment, whereas the latter discharges into the moderator and once the calandria rupture discs burst, into containment. Such an inlet feeder break scenario is called a “stagnation” break. For an inlet feeder break which is slightly larger or smaller than that for the stagnation break case, the result is a channel flow which is low enough to result in fuel failure but high enough that the pressure tube remains intact for a relatively extended period of time. This event is identified as the “off-stagnation” break.

2. Event Description

The initiating event is considered to be a spontaneous break in an inlet feeder. Stagnation breaks are defined as breaks which result in near-zero channel flow. This class of breaks is characterized by rapid fuel and pressure tube heat-up and channel failure.

3. Analysis Methodology

The objective of the fuel analysis is to estimate the quantity and timing of fission product release from the fuel in the affected channel. Fuel behaviour in the unaffected channels is similar to that for a small loss-of-coolant accident. Therefore, no fuel failures, and hence no additional fission product releases are expected from the fuel in the unaffected channels. Fuel thermal and mechanical behaviour depends on the coolant conditions, the power transient and the duration of the transient. One type of break is considered in this analysis. “Stagnation breaks” are defined as breaks which result in near-zero channel flow. This class of breaks is characterized by rapid fuel and pressure tube heat-up and channel failure.

When a fuel sheath fails, the gap inventory of radionuclides is released over a period of time as the fission gases within the fuel element migrate towards the defect site. A fraction of the gap inventory of some isotopes is retained on the inside surface of the sheath. The fuel heatup due to coolant degradation and power increase has the effect of enhancing the migration of fission gases from within grains to the grain boundary, from where they are subsequently available for release. Chemical interactions between the Zircaloy cladding and the UO_2 fuel pellets results in additional release of fission products. The extent of the interaction depends on the temperatures of the sheath and of the interface, the fuel/sheath interfacial contact pressure, the oxygen content of the sheath and time of interaction. Releases due to this interaction may be significant at high fuel and sheath temperatures as described in reference [1]. Exposure of the fuel pellets to super-heated coolant following sheath failure causes oxidation of UO_2 which enhances both fission product diffusion through grains and grain growth [2]. Fuel rewet following the channel failure or injection of emergency core coolant can result in fuel pellet cracking and powdering due to induced thermal stresses. Some of the fission gas which is stored on grain boundaries will be released when cracking or powdering occurs.

A feeder break can be postulated to occur in any of the 380 channels in the reactor at any time in the reactor’s operating history. The objective of the fuel analysis for each single channel event is to conservatively estimate the fission product releases from fuel in the affected channel.

It is impractical to analyze each of the 380 channels for each single channel event. Instead, a high-powered 'limiting' channel is constructed such that releases are overestimated relative to any "real" channel. This limiting channel is O6_mod having a channel power of 7.3 MW, with two central bundles at 935 kW [3 and 4].

3.1 Conversion of Bundle Powers to Element Linear Powers

The power density of a fuel bundle decreases radially towards its centre because of the attenuation of thermal neutrons by the bundle materials. This attenuation varies with time because of preferential burnup. The peak bundle power occurs at a bundle average burnup of ≈ 60 MW·h/kg(U) (also known as the plutonium peak). Table 2 gives the relative element powers at the plutonium peak.

The element power normalized to the bundle power (Column 3) in and Table 2 are used to calculate element powers from bundle powers by the following equation:

$$P_{element} = \frac{P_{bundle} \text{ x RPF}}{N_{element} \text{ x L}} \qquad (1)$$

where,
$P_{element}$ is the element linear power in kW/m
P_{bundle} is the bundle power in kW
RPF is the element radial power factor from Table 2
$N_{element}$ is the number of fuel elements per bundle (43), and
L is the fuel stack length in metres (L = 0.4823 m and 0.4805 m for large element and small element respectively)
For example, the outer element linear power for a 935 kW bundle at plutonium peak is:

$$P_{element} = \frac{935 \text{ x } 1.058}{43 \text{ x } 0.4805} = 47.88 \ \frac{\text{kW}}{\text{m}} \qquad (2)$$

The burnup of a single element also varies with the radial position of the element within the bundle. Table 3 gives the conversion factors for burnup. For example, if the bundle burnup is 2400 MW·h, then the outer element burnup is equal to:

$$B_{element} = \frac{B_{bundle} \text{ x RBF}}{m_{(U)}} = \frac{2400 \text{ x } 1.1342}{19.2} = 141.8 \frac{\text{MW h}}{\text{kg(U)}} \qquad (3)$$

where,
$B_{element}$ is the element burnup in MW·h/kg(U)
B_{bundle} is the bundle burnup in MW·h
$m_{(U)}$ is the nominal mass of uranium in the bundle in kg
RBF is the element radial burnup factor from Table 3.
If the bundle average burnup (B_{avg}) is 125 MW·h/kg(U), then the outer element burnup is equal to:

$$B_{element} = B_{avg} \text{ x RBF} = 125 \text{ x } 1.1342 = 141.8 \ \frac{\text{MW} \bullet \text{h}}{\text{kg(U)}} \qquad (4)$$

3.2 Methodology To Estimate Fission Product Releases From The Fuel

Fission product release calculations consist of three parts. First, the fission product inventory in the core is estimated using the ELESTRES [5] computer code to simulate the operating history of the fuel elements in the channel. Secondly, based on the fuel temperatures following the accident, the fractional release of the different chemical species is estimated. Finally, the release of different isotopes in the channel is determined by multiplying the fractional releases in the fuel.

Upper Bound Approach for Fission Product Inventory Calculations

The radionuclide (i.e., fission product) inventory and distribution within the fuel during normal operation is the starting point for this analysis. The factors affecting the fission product inventory are the fuel power, and burnup at the time of the accident and, to a lesser extent, its power/burnup history. The fission products are created initially within the UO_2 matrix. They can migrate by thermal or irradiation diffusion processes. This redistributes the fission gases within the grains of the fuel pellet, some of which migrate to grain boundaries. The fission products at the grain boundaries can also migrate out of the fuel pellet to the gaps between the UO_2 fuel pellets and the sheath, as well as to the cracks within the fuel pellets. The ease with which the fission products may be released from the fuel element during an accident depends on their location at that time. Thus, there are three main sources of radionuclides that must be considered:

a) the grain-bound or matrix inventory
b) the grain boundary inventory
c) the gap inventory

Each of these sources contributes to the release during the accident to a varying degree, the gap inventory being the easiest to release, and the matrix inventory the most difficult. The calculation of the initial fission product distribution is performed with the ELESTRES computer code. The fission product release model is based on the American Nuclear Society (ANS) Standard 5.4 [6]. The ELESTRES analysis provides the fuel temperature and distribution of the various kinds of fission products within a fuel element. This information is used with the ANS 5.4 model for a release estimate to the gap. The ANS 5.4 model is basically a Booth diffusion-type model [7] which is empirically fitted to experimental data. The gap fission product inventory predicted using the ANS 5.4 has been compared with experiments on CANDU fuel. It has been shown that the ANS 5.4 model over-predicted the steady-state release of noble gases by several orders of magnitude [8 and 9]. This is because the ANS 5.4 model parameters are fitted to predominantly light water reactor fuel data at low power and high burnup. To account for this over-estimation, the free inventory has been reduced to 20% of the ANS 5.4 value, and the grain boundary inventory has been correspondingly increased.

In the ELESTRES code, both the total and gap inventories are calculated using the ANS 5.4 model. It also calculates the total number of fission gas atoms in the element, the number of atoms on the grain boundary and the number of atoms in the gap. These calculations are independent of the inventory calculations. Therefore, the grain boundary inventories are estimated by assuming that the ratio of the grain boundary inventory to the total inventory of each isotope was equal to the ratio of the number of fission gas atoms on the grain boundary to the total number of fission gas atoms in the element. This latter ratio is calculated in ELESTRES. Note that the grain boundary inventory which results from this calculation does not include the effect of decay during diffusion which occurs and is accounted for with the ANS 5.4 gap model. Hence, this inventory is over-estimated.

The power and burnup distribution in the high powered limiting channel O6_mod is provided in Table 1. The limiting burnup distribution is determined by searching through the data provided by the time-averaged fuel management study provided in reference [3]. The data was searched for the maximum bundle average burnup for each bundle location accounting for the fact that there are two different fuelling directions for the channels in the reactor.

An element from each ring of the 12 bundles in the fuel string was simulated using the ELESTRES code. The power/burnup histories for these runs were derived from the limiting power envelope as follows:

The reference power envelope is a curve of bundle power versus bundle average burnup which encompasses most of the bundle powers predicted in a fuel management simulation of reactor operation from startup until the time that the last remaining bundle from the core is discharged. For safety analysis, the limiting power envelope is derived by modifying the reference overpower envelope such that the maximum power is equal to the limiting condition for bundle powers. The limiting power envelope for fuel elements in different rings is given in Table 4 and Figure 1. Table 4 provides the assumed bounding power and burnup for fuel elements in different bundles of the channel at the time of the accident. These power/burnup points are compared to the limiting power envelope. If the power/burnup point of the element is above the limiting power envelope then the element is assumed to have operated on the limiting power envelope itself. Likewise, if the element is in the intermediate ring and the power/burnup point is above the limiting power envelope for intermediate elements, then it is also assumed to operate on the intermediate element limiting power envelope. Similar methodology is used for the centre elements and the elements in the inner rings. At the time of the accident, the element power is instantaneously boosted to coincide with the element power given in Table 4. Such a combination of high power and high burnup results in an upper bound fission product inventory prediction. It should be noted that a more realistic approach is taken to estimate the fuel element thermal-mechanical condition at the time of the accident.

If the power/burnup point of a fuel element at the time of the accident is below the limiting envelope, the burnup history for that element is constructed as follows. The element is assumed to operate parallel to the limiting power envelope at a power ΔP lower than the limiting power envelope (Figure 2). ΔP is the difference in power between the limiting power (from the envelope) and the element power at the element burnup at the time of the accident. This process was repeated for each element that operated below the limiting power envelope. The power of each of the 48 simulated elements is boosted by 5% to account for the increase in channel power due to channel coolant voiding. The 5% power increase of all elements is assumed to last for 15 minutes (Figure 2). The complete power histories of each of the 48 simulated elements is calculated using the methodology discussed above.

To further bound the channel fission product inventories, each ELESTRES output was scanned and the maximum total inventory and the corresponding gap inventory of each isotope was recorded. This approach results in an over-prediction of the overall channel fission product inventory. The inventories within the individual grains were calculated using the total, gap and grain boundary inventories provided by the ELESTRES computer code.

Release Calculations - Stagnation Breaks

Following a feeder stagnation break, the fission product release from failed fuel elements in the affected channel is rapid and substantial. In order to simplify the calculations, the following assumptions are made:

a) The fuel geometry is assumed unchanged in the CATHENA simulations.
b) For the purpose of estimating fission product releases, all fuel sheaths in the channel are assumed to fail at the beginning of the accident and the entire gap inventory is assumed to be released instantaneously. No credit is taken for the diffusion time of fission gas to the defect site, nor for retention factor of any radionuclide on the sheath inside surface.

The result of these assumptions is that the fractional release of radionuclides at time zero is over-estimated. Moreover, by assuming gap and grain boundary release at the beginning of the transient, the timing of releases is conservatively underestimated.
The calculation of transient fission product release from the fuel grains and grain boundary is performed with Gehl's release model [10]. This model is empirically based on unconstrained fuel, and over-predicts the release because the external coolant pressure would be expected to provide constraint. Gehl's model correlates the percentage of fission gas release (Zc) with the centreline fuel temperature ($T_{c/1}$) in K, and the time-averaged centreline heating rate $\left(\frac{dT_{c/l}}{dt}\right)$ in K/s as follows:

$$Zc \quad = \quad 7.58 \times 10^{-19}\ T_{c/l}^{5.7} \left(\frac{dT_{c/1}}{dt}\right)^{-0.346} \qquad \text{(A.II.9.1)}$$

Additional releases are superimposed on the transient releases predicted using Gehl's model, to account for Zircaloy/UO_2 interaction and UO_2 oxidation. These releases are temperature dependent and are given in Table 5 as a percentage release of fission products located within the grains of fuel and the grain boundary. They are based on estimates of the amount of UO_2 which can theoretically be dissolved by Zircaloy. These estimates are in turn based on the Zircaloy/UO_2 phase diagram. The additional release fractions are added to the releases predicted by Gehl's model when the fuel volume average temperature predicted by CATHENA reaches the temperature specified in Table 5.

Fuel rewet following the channel failure or injection of emergency core coolant can result in fuel pellet cracking and powdering due to induced thermal stresses. Therefore, the remaining fission gas which is stored on the grain boundaries is assumed to be released at the time of channel failure (11.4 seconds). The percentage releases are calculated for an element from each ring of each bundle in the channel using the CATHENA-predicted temperature transients. The percentages are assumed to apply to all of the isotopes modelled in the ELESTRES code. The total activity release is calculated at appropriate time intervals and summed over all of the elements in the channel.

4. Analysis Results

Fission Product Release Estimates For Fuel

For each bundle in channel 06_mod the inventories for each of the 30 considered nuclides, are shown in Table 6 to Table 9. These tables provide the total inventories, the gap inventories, the grain boundary inventories and the grain bound inventories.

Feeder Stagnation Break Results

The channel was predicted to fail at 11.4 seconds following the accident. To ensure that the releases are not underpredicted, the transient releases were calculated based on fuel heatup of 13.4 seconds. For these two additional seconds, the fuel cooling effect of the channel rupture is not taken into consideration. The total channel release at 13.4 seconds after the accident is calculated to be 67285 TBq, i.e. approximately 25% of the total inventory. Release of each individual nuclide is given in Table 10. The release calculations were based on CATHENA fuel temperature predictions.

5. Conclusions

The stagnation break is chosen for the analysis of the consequences of inlet feeder breaks. The behaviour associated with the inlet feeder stagnation break is near-stagnant channel flow, rapid fuel heatup and failure of some of the fuel elements, followed by early pressure tube/calandria tube failure. For the stagnation break event, the time of pressure tube/calandria tube failure is 11.4 seconds for channel O6_mod. The total channel fission product release at 13.4 seconds after initiation of the accident is calculated to be about 67285 TBq. This value is quite lower than that of 37 standard element analysis (96922 TBq) and the releases of radionuclides to the environment are limited such that public doses are below the acceptable limits.

6. References

1. P. Hoffmann, D. Kerwin-Peck and P. Nikolopoulos, "Physical and Chemical Phenomena Associated with the dissolution of Solid UO_2 by Molten Zircaloy-4", *Zirconium in the Nuclear Industry: Sixth International Symposium*, ASTM STP 824, D.G. Franklin and R.B. Adamson, Eds., American Society for Testing and Materials, 1984, pp. 810-834.
2. D.S. Cox, F.C. Iglesias, C.E.L. Hunt, N.A. Keller, R.D. Barrand, J.R. Mitchell, R.F. O'Connor, "Oxidation of UO_2 in Air and Steam with Relevance to Fission Product Release", *Proceedings of the symposium on Chemical Phenomena Associated with Radioactivity Releases During Severe Nuclear Plant Accidents*, Anaheim, California, U.S.A., September 9-12, 1986. Report NUREG/CP-0078.
3. M. Shad and H. Chow, "Fuel Management Report", 98-03310-AR-003, Revision 0, 1991 December.
4. A. Ranger, "CATHENA Slave Channel Models for Wolsong2 Trip Coverage Analysis", 98-03500-AR-005, Revision 0, March 1992.
5. M. Tayal, "Modelling CANDU Fuel Under Normal Operating Conditions: ELESTRES Code Description", AECL-9331, 1987, "Users' Manual for the M11C Version of the ELESTRES Code", TTR-234A, February 1989.
6. American National Standard Method for Calculation the Fractional Release of Volatile Fission Products from Oxide Fuel, ANSI/ANS-5.4-1982.
7. A.H. Booth, "A Method of Calculating Fission Gas Diffusion from UO_2 Fuel and its Application to the X-2-f Loop Test", AECL-496 (1957) or CRDC-721 (1957).
8. B.J. Lewis, C.E.L. Hunt and F.C. Iglesias, "Source Term of Iodine and Noble Gas Fission Products in the Fuel-to-Sheath Gap of Intact Operating Nuclear Fuel Elements", Journal of Nuclear Materials, 172 (1990) 197-205.
9. B.J. Lewis, F.C. Iglesias, D.S. Cox and E. Gheorghiu, "A Model for Fission Gas Release and Fuel Oxidation Behaviour for Defected Fuel Elements", *Nuclear Technology,* Vol. 92, December 1990.
10. S.M. Gehl, "The Release of Fission Gas during Transient Heating of LWR Fuel", Argonne National Lab Report ANL-80-108, March, 1981.

Table 1Table .II.8-1
Bundle Power and Burnups at the Time of Refuelling in the Limiting Channel used in Fission Product Relaese Analysis for Single Channel Events (Channel Inlet = Position 1)

Position	Power (kW)	Channel*	Bundle Average Burnup (MW·h/kg(U))
1	111.7	M11	43.8
2	406.1	M11	102.4
3	619.7	M9	139.8
4	761.4	M15	161.5
5	874.0	J17	175.6
6	935.0	J6	188.5
7	935.0	J6	188.5
8	875.6	J6	175.6
9	744.9	M14	198.6
10	577.5	M12	229.9
11	363.8	M11	230.0
12	95.3	M9	198.6

* Channel having the maximum bundle average burnup in the respective bundle position just before refuelling as calculated in the time-average fuel management study [3].

Table 2Table .II.8-2
Bundle Power Distribution at Plutonium Peak (≈ 60 MW·h/kg(U))

Element ring	Number of elements	Element Power	
		Normalized to bundle average*	Normalized to outer element
Outer	21	1.058	1.000
Intermediate	14	0.8707	0.8230
Inner	7	1.080	1.0208
Centre	1	1.0325	0.9759

* Also known as the Radial Power Factor (RPF)

Table 3Table .II.8-3
Burnup Conversion Ratios

Element ring	Number of Elements	Relative element burnup* (Bundle average = 1.000)	Relative element burnup (Normalized to outer elements)
Outer	21	1.1342	1.000
Intermediate	14	0.9377	0.6987
Inner	7	0.8309	0.6191
Centre	1	0.7941	0.5917

* Radial Burnup Factor (RBF)

Table 4Table .II.8-4
Limiting Power Envelopes forEach Ring of Elements

Outer Elements		Intermediate Elements		Inner Elements		Centre Element	
Power (kw/m)	Burnup (MWh/kg (U))	Power (kw/m)	Burnup (MWh/kg (U))	Power (kw/m)	Bunup (MWh/kg (U))	Power (kw/m)	Burnup (MWh/kg (U))
47.1	11.32	38.96	9.39	48.1	8.33	45.98	7.96
47.55	22.65	39.23	18.77	48.44	16.64	46.3	15.9
47.82	34	39.36	28.14	48.62	24.94	46.46	23.84
47.88	45.36	39.41	37.5	48.74	33.23	46.6	31.76
47.8	56.73	39.35	46.85	48.73	41.52	46.61	39.68
47.6	68.09	39.21	56.2	48.62	49.81	46.52	47.61
47.28	79.44	39	65.55	48.45	58.12	46.39	55.56
46.87	90.79	38.73	74.9	48.21	66.44	46.19	63.52
46.4	102.14	38.4	84.25	47.9	74.76	45.92	71.49
45.88	113.46	38.04	93.61	47.54	83.13	45.6	79.51
45.32	124.77	37.64	102.97	47.13	91.51	45.25	87.54
44.73	136.08	37.21	112.33	46.7	99.89	44.86	95.58
44.13	147.35	36.76	121.7	46.22	108.32	44.43	103.68
43.52	158.62	36.3	131.08	45.73	116.77	43.99	111.79
42.91	169.88	35.83	140.45	45.22	125.21	43.53	119.9
42.3	181.12	35.36	149.83	44.69	133.7	43.06	128.07
41.79	192.35	34.89	159.21	44.16	142.19	42.58	136.25
41.11	203.58	34.43	168.59	43.65	150.69	42.11	144.44
40.53	214.8	33.97	177.98	43.12	159.22	41.63	152.66
39.98	226	33.52	187.37	42.6	167.76	41.15	160.91
39.45	237.2	33.08	196.75	42.1	176.31	40.69	169.15
38.94	248.39	32.66	206.14	41.6	184.87	40.22	177.42
38.45	259.58	32.24	215.52	41.11	193.44	39.77	185.71
37.99	270.77	31.84	224.91	40.64	202.01	39.33	194
37.55	281.95	31.46	234.29	40.18	210.6	38.89	202.3
37.14	293.14	31.08	243.66	39.72	219.18	38.46	210.61

Table 5Table .II.8-5
Release Criteria Applied in Addition to Gehl’s Model to Account for Interaction Between Zircaly and Uranium Dioxide

Temperature (°C)	Inventory Release (percent)	
	Free	Bound
800	100	0
1800	100	10
2100	100	20
2400	100	100

Table 6Table .II.8-14

Total Inventories per Bundles (TBq)

ISOTOPE	BUNDLE 1	BUNDLE 2	BUNDLE 3	BUNDLE 4	BUNDLE 5	BUNDLE 6	BUNDLE 7	BUNDLE 8	BUNDLE 9	BUNDLE 10	BUNDLE 11	BUNDLE 12	CHANNEL
I-131	105.97	416.59	651.21	800.34	863.95	863.95	863.95	863.95	821.59	700.41	522.44	223.38	7697.71
I-132	162.86	641.94	1009.67	1250.51	1356.92	1356.92	1356.92	1356.92	1285.80	1088.01	807.12	343.24	12016.79
I-133	254.35	1003.22	1577.89	1954.28	2120.57	2120.57	2120.57	2120.57	2009.42	1700.32	1261.36	536.41	18779.54
I-135	238.87	942.01	1481.62	1835.05	1991.20	1991.20	1991.20	1991.20	1886.83	1596.59	1184.40	503.68	17633.84
I-137	131.81	497.13	781.90	968.42	1050.82	1103.03	1103.03	1050.82	995.74	842.57	625.05	265.81	9416.14
KR-83M	19.70	77.63	122.10	151.22	164.09	164.09	164.09	164.09	155.49	131.57	97.61	41.51	1453.20
KR-85M	48.08	189.60	298.20	369.34	400.76	400.76	400.76	400.76	379.76	321.34	238.38	101.37	3549.13
KR-85	0.49	1.16	1.59	1.84	2.00	2.14	2.14	2.00	2.25	2.59	2.56	2.10	22.86
KR-87	93.71	368.74	579.97	718.31	779.44	779.44	779.44	779.44	738.58	624.97	463.63	197.16	6902.83
KR-88	132.16	521.02	819.47	1014.95	1101.31	1101.31	1101.31	1101.31	1043.59	883.06	655.08	278.58	9753.15
KR-89	178.99	676.28	1063.67	1317.40	1429.49	1497.00	1497.00	1429.49	1354.57	1146.20	850.29	361.60	12801.98
XE-133M	7.19	28.36	44.61	55.25	59.94	59.94	59.94	59.94	56.80	48.07	35.66	15.17	530.88
XE-133	234.27	923.49	1449.44	1790.34	1941.56	1941.56	1941.56	1941.56	1840.57	1560.88	1159.87	494.04	17219.16
XE-135M	41.71	161.23	253.59	314.08	340.81	340.81	340.81	340.81	322.94	273.27	202.72	86.21	3018.98
XE-135	27.25	107.49	169.06	209.39	227.20	227.20	227.20	227.20	215.30	182.18	135.15	57.47	2012.10
XE-137	241.48	913.65	1437.01	1779.79	1931.24	2017.99	2017.99	1931.24	1830.02	1548.51	1148.74	488.51	17286.17
XE-138	238.61	921.11	1448.75	1794.33	1947.02	1947.02	1947.02	1947.02	1844.97	1561.16	1158.13	492.50	17247.63
I-134	286.02	1122.65	1765.74	2186.93	2373.03	2373.03	2373.03	2373.03	2248.65	1902.75	1411.53	600.26	21016.63

Table 7Table .II.8-15

Gap Inventories per Bundle (TBq)

ISOTOPE	BUNDLE 1	BUNDLE 2	BUNDLE 3	BUNDLE 4	BUNDLE 5	BUNDLE 6	BUNDLE 7	BUNDLE 8	BUNDLE 9	BUNDLE 10	BUNDLE 11	BUNDLE 12	CHANNEL
I-131	0.00	0.00	0.07	0.94	2.81	2.81	2.81	2.81	1.40	0.18	0.01	0.00	13.83
I-132	0.00	0.00	0.16	1.96	5.09	5.09	5.09	5.09	2.77	0.39	0.01	0.00	25.64
I-133	0.00	0.00	0.06	0.72	1.90	1.90	1.90	1.90	1.02	0.14	0.01	0.00	9.56
I-135	0.00	0.00	0.03	0.38	1.01	1.01	1.01	1.01	0.54	0.08	0.00	0.00	5.08
I-137	0.00	0.00	0.00	0.01	0.02	0.12	0.12	0.02	0.01	0.00	0.00	0.00	0.30
KR-83M	0.00	0.00	0.00	0.01	0.02	0.02	0.02	0.02	0.01	0.00	0.00	0.00	0.08
KR-85M	0.00	0.00	0.00	0.02	0.06	0.06	0.06	0.06	0.03	0.00	0.00	0.00	0.32
KR-85	0.00	0.00	0.00	0.01	0.02	0.03	0.03	0.03	0.01	0.00	0.00	0.00	0.14
KR-87	0.00	0.00	0.00	0.02	0.07	0.07	0.07	0.07	0.04	0.00	0.00	0.00	0.33
KR-88	0.00	0.00	0.00	0.05	0.14	0.14	0.14	0.14	0.07	0.01	0.00	0.00	0.70
KR-89	0.00	0.00	0.00	0.01	0.02	0.17	0.17	0.02	0.01	0.00	0.00	0.00	0.42
XE-133M	0.00	0.00	0.00	0.01	0.03	0.03	0.03	0.03	0.02	0.00	0.00	0.00	0.17
XE-133	0.00	0.00	0.10	1.32	4.18	4.18	4.18	4.18	2.06	0.25	0.01	0.00	20.48
XE-135M	0.00	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.06
XE-135	0.00	0.00	0.00	0.06	0.17	0.17	0.17	0.17	0.09	0.01	0.00	0.00	0.84
XE-137	0.00	0.00	0.00	0.01	0.04	0.26	0.26	0.04	0.02	0.00	0.00	0.00	0.62
XE-138	0.00	0.00	0.00	0.03	0.07	0.07	0.07	0.07	0.04	0.01	0.00	0.00	0.36
I-134	0.00	0.00	0.01	0.17	0.44	0.44	0.44	0.44	0.24	0.03	0.00	0.00	2.21

Table 8Table .II.8-16

Grain Boundary Inventories per Bundle (TBq)

ISOTOPE	BUNDLE 1	BUNDLE 2	BUNDLE 3	BUNDLE 4	BUNDLE 5	BUNDLE 6	BUNDLE 7	BUNDLE 8	BUNDLE 9	BUNDLE 10	BUNDLE 11	BUNDLE 12	CHANNEL
I-131	3.53	21.05	40.44	96.18	139.11	139.28	139.28	140.49	107.13	57.10	39.63	16.36	939.58
I-132	5.43	32.44	62.90	152.26	221.19	221.46	221.46	223.37	169.98	89.18	61.25	25.13	1486.06
I-133	8.48	50.69	97.53	228.56	321.49	321.92	321.92	324.89	252.42	137.49	95.66	39.28	2200.34
I-135	7.96	47.60	91.48	213.44	298.77	299.17	299.17	301.96	235.35	128.87	89.81	36.88	2050.48
I-137	4.39	25.12	48.22	111.86	155.61	163.97	163.97	157.29	123.10	67.86	47.39	19.46	1088.24
KR-83M	0.66	3.92	7.53	17.49	24.36	24.39	24.39	24.62	19.25	10.60	7.40	3.04	167.64
KR-85M	1.60	9.58	18.40	42.75	59.57	59.65	59.65	60.22	47.07	25.90	18.07	7.42	409.89
KR-85	0.02	0.06	0.10	0.21	0.30	0.32	0.32	0.30	0.28	0.21	0.19	0.15	2.45
KR-87	3.12	18.63	35.77	83.05	115.63	115.79	115.79	116.88	91.42	50.35	35.15	14.44	796.03
KR-88	4.40	26.33	50.55	117.42	163.57	163.79	163.79	165.33	129.27	71.15	49.67	20.40	1125.65
KR-89	5.96	34.17	65.59	152.17	211.69	222.57	222.57	213.98	167.46	92.31	64.47	26.48	1479.41
XE-133M	0.24	1.43	2.75	6.43	9.01	9.02	9.02	9.10	7.09	3.88	2.70	1.11	61.79
XE-133	7.81	46.66	89.79	212.02	304.12	304.51	304.51	307.22	235.69	126.70	87.97	36.18	2063.18
XE-135M	1.39	8.15	15.64	36.29	50.50	50.57	50.57	51.04	39.94	22.01	15.37	6.31	347.77
XE-135	0.91	5.43	10.44	24.43	34.30	34.34	34.34	34.66	26.97	14.72	10.25	4.21	235.00
XE-137	8.05	46.16	88.61	205.59	286.00	300.11	300.11	289.09	226.25	124.71	87.10	35.77	1997.56
XE-138	7.95	46.54	89.34	207.32	288.47	288.87	288.87	291.59	228.17	125.74	87.81	36.06	1986.73
I-134	9.53	56.72	108.93	253.21	353.01	353.49	353.49	356.81	278.85	153.36	107.03	43.95	2428.39

Table 9Table .II.8-17

Grain Bound Inventories per Bundle (TBq)

ISOTOPE	BUNDLE 1	BUNDLE 2	BUNDLE 3	BUNDLE 4	BUNDLE 5	BUNDLE 6	BUNDLE 7	BUNDLE 8	BUNDLE 9	BUNDLE 10	BUNDLE 11	BUNDLE 12	CHANNEL
I-131	102.43	395.54	610.69	703.23	722.03	721.86	721.86	720.65	713.06	643.13	482.80	207.02	6744.30
I-132	157.43	609.50	946.60	1096.28	1130.64	1130.37	1130.37	1128.46	1113.05	998.43	745.86	318.10	10505.09
I-133	245.87	952.53	1480.31	1724.99	1797.17	1796.74	1796.74	1793.77	1755.98	1562.69	1165.70	497.13	16569.63
I-135	230.91	894.41	1390.11	1621.23	1691.41	1691.01	1691.01	1688.22	1650.94	1467.64	1094.59	466.80	15578.28
I-137	127.42	472.01	733.69	856.55	895.20	938.93	938.93	893.51	872.64	774.71	577.66	246.35	8327.60
KR-83M	19.04	73.71	114.57	133.73	139.72	139.69	139.69	139.46	136.23	120.97	90.20	38.47	1285.47
KR-85M	46.48	180.02	279.81	326.57	341.13	341.05	341.05	340.48	332.65	295.44	220.31	93.95	3138.92
KR-85	0.48	1.10	1.49	1.62	1.68	1.79	1.79	1.67	1.96	2.38	2.37	1.95	20.28
KR-87	90.58	350.11	544.20	635.24	663.74	663.58	663.58	662.49	647.13	574.62	428.47	182.72	6106.48
KR-88	127.76	494.69	768.92	897.48	937.61	937.38	937.38	935.84	914.24	811.89	605.41	258.18	8626.80
KR-89	173.02	642.11	998.08	1165.22	1217.78	1274.25	1274.25	1215.49	1187.10	1053.89	785.83	335.12	11322.14
XE-133M	6.95	26.93	41.85	48.80	50.90	50.89	50.89	50.81	49.70	44.19	32.96	14.06	468.93
XE-133	226.46	876.82	1359.55	1577.00	1633.26	1632.87	1632.87	1630.15	1602.82	1433.93	1071.89	457.87	15135.50
XE-135M	40.32	153.09	237.95	277.79	290.30	290.23	290.23	289.75	283.00	251.26	187.35	79.90	2671.15
XE-135	26.35	102.06	158.61	184.89	192.74	192.69	192.69	192.38	188.24	167.45	124.90	53.26	1776.26
XE-137	233.43	867.48	1348.40	1574.19	1645.20	1717.62	1717.62	1642.11	1603.75	1423.80	1061.64	452.74	15287.99
XE-138	230.66	874.57	1359.41	1586.99	1658.47	1658.08	1658.08	1655.35	1616.76	1435.42	1070.32	456.44	15260.55
I-134	276.49	1065.93	1656.79	1933.55	2019.57	2019.10	2019.10	2015.77	1969.56	1749.36	1304.50	556.31	18586.03

Table 10Table .II.8-18

Cumulative Fission Product Releases at 13.4 Seconds

Isotope	Release (TBq) CANFLEX-NU	Release (TBq) 37 Standard
Cs-137	42.9	65.5
Cs-138	4552.0	6426.5
I-131	1981.7	2996.4
I-132	3121.4	4666.2
I-133	4761.2	6871.1
I-134	5295.6	7502.1
I-135	4455.8	6370.9
I-137	2387.8	3377.2
Kr-83m	365.9	517.0
Kr-85m	894.1	1265.6
Kr-85	5.1	7.7
Kr-87	1737.9	2454.0
Kr-88	2456.4	3473.2
Kr-89	3245.2	4582.1
Sr-89	2850.3	4351.9
Sr-90	42.2	64.1
Te-131m	257.9	379.4
Te-131	1797.0	2551.9
Te-132	3095.3	4589.8
Te-133m	2052.9	2926.2
Te-133	2749.1	3894.0
Te-135	2424.5	3430.0
Xe-133m	134.2	192.1
Xe-133	4405.3	6538.5
Xe-135m	759.6	1071.0
Xe-135	509.4	733.3
Xe-137	4380.6	6178.4
Xe-138	4339.5	6118.2
Ru-103	2103.8	3204.5
Ru-106	81.0	123.4
Total	67285.5	96922.2

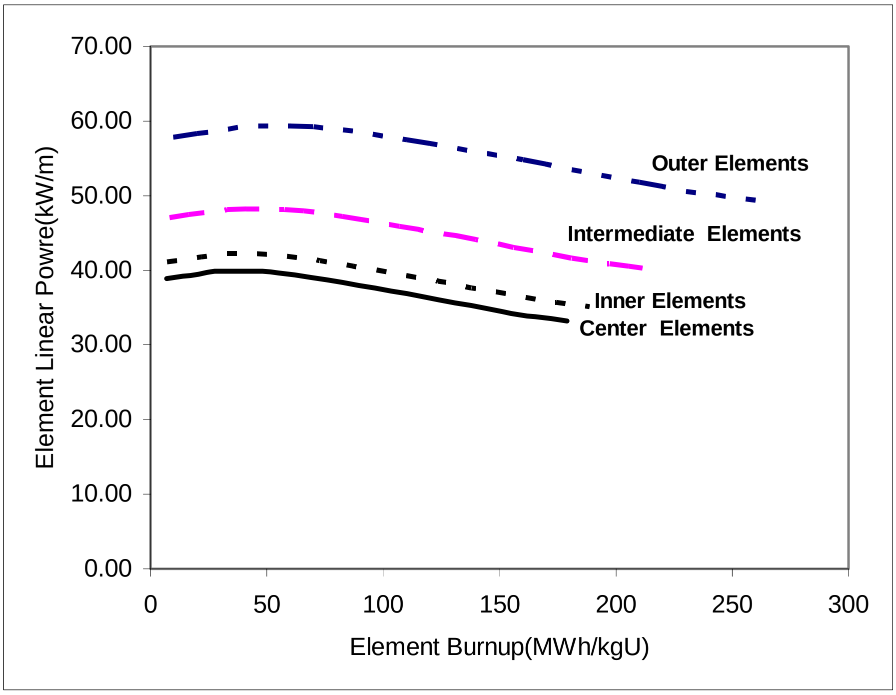

Figure 1. Limiting Power Envelopes for Each Ring of Elements

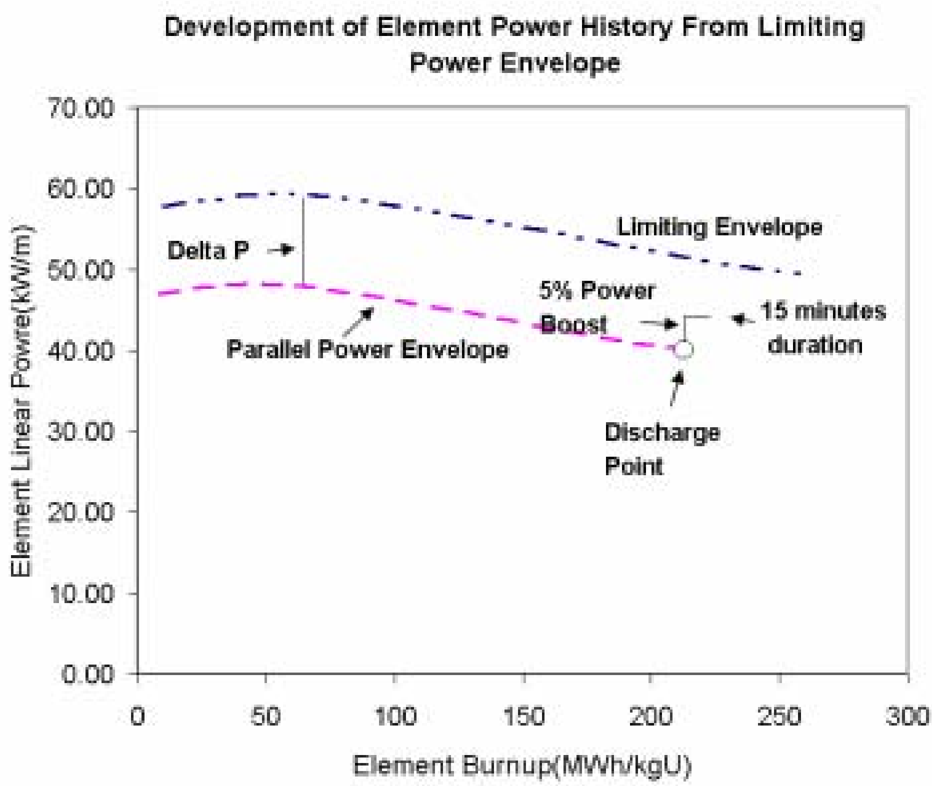

Figure 2. Development of Element Power History from Limiting Power Envelope