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Fission Product Release Assessment for the Inlet Feeder Break Accident for Wolsong1 NPP Loaded with CANFLEX-NU Fuel

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ABSTRACT

The fission product release assessment for the inlet feeder 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. 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 the stagnation break event, the break area was 17.75 cm^2 and the time of pressure tube/calandria tube contact failure is 11.4 seconds for channel O6_mod. The total channel fission product release 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). For the off-stagnation break event, the 37 cm^2 break area results are taken to represent the limiting off-stagnation break and total release from the channel is predicted to be 77064 TBq, 27% of the total inventory. The fission product release of CANFLEX-NU fuel is lower than that of 37 standard element analysis (89925 TBq, 34% of the total inventory). The initial fission product release and fuel temperatures of the CANFLEX-NU fuel in general are lower than 37 element fuel, which leads to lower oxidation releases, and the elements fail later in the transient. These factors contribute to the overall reduction in fission product release from CANFLEX-NU fuel compared with 37 element fuel. In terms of dose consequences, those releases of radionuclides to the environment are limited such that public doses are below the acceptable limits.

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. Off-stagnation breaks causing reverse flow are more limiting than those forward flow, because there is a direct path to the break for radionuclide releases into containment.

2. CANFLEX-NU Fuel Description

The CANFLEX-NU (CANDU-FLEXIBLE-Natural Uranium) fuel bundle is a 43-element natural uranium design containing an array of fuel elements of two different diameters. Figure 1 shows a schematic diagram of a CANFLEX-NU bundle cross section. The larger (inner 8) elements have a fuel pellet diameter of 12.67 mm, and the smaller (outer 35) elements have a 10.73 mm fuel pellet diameter. Different element diameters and larger number of elements in CANFLEX-NU bundle design is well known to reduce the peak linear outer element power ratings by approximately 20% at the same overall bundle power output (Alavi, 1995).

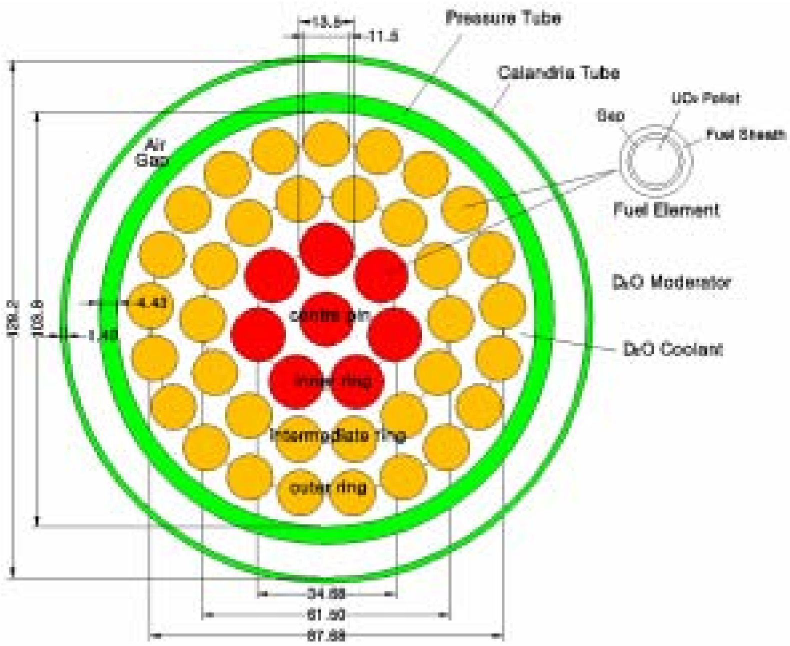

Figure 1. Cross Section of the CANFLEX Fuel Channel

3. Fission Product Release Mechanism

Fuel thermal and mechanical behaviour depends on the coolant conditions, the power transient and the duration of the transient. Two types of break, Stagnation and Off-stagnation, are considered in this analysis. 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 (Hoffmann et al., 1984).

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 (Cox et al, 1986). In hyperstoichiometric UO_{2+x}, fission gas diffuses through the grains to the grain boundary at an increased rate. This is explained by the observation that one of the factors affecting the diffusion of gas atoms and bubbles is the concentration of vacancies (or defects) within the UO_2 lattice (Killeen and Turnbull, 1987). The number of these vacancies increases due to damage caused during irradiation and because of the excess oxygen which is incorporated into the existing lattice when the fuel becomes oxidized. Hyperstoichiometry reduces the number of oxygen vacancies and increases the number of uranium vacancies. This change in the composition of the fuel enhances the mobility of both gas atoms and inter-granular bubbles (Cox et al, 1986). UO_2 oxidation in steam has also been reported to cause accelerated grain growth which leads to enhanced release of fission products by grain boundary sweeping (Bittel et al). 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.

4. Analysis Methodology

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 obtained from Table A-3 and 7 of the Wolsong NPP Safety Report (shown in Table 1). This limiting channel is O6_mod having a channel power of 7.3 MW, with two central bundles at 935 kW (Ranger, 1992).

Table 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 (Shad and Chow, 1991).

4.1 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 (Tayal, 1989) 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.

4.1.1 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 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. 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 (Booth, 1957) 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 (Lewise et al, 1990). 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.

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
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 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.

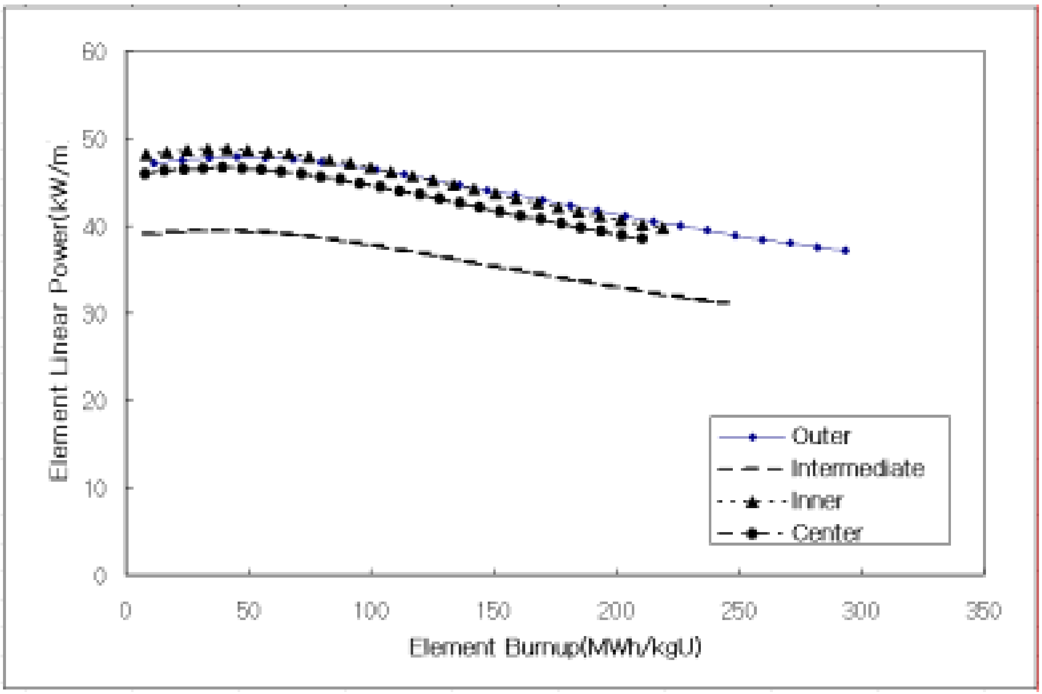

Figure 1. Limiting Power Envelopes for Each Ring of Elements

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.

The thermalhydraulic analysis of a feeder break was performed using the one-dimensional, two-fluid thermalhydraulic computer code CATHENA (Hanna, 1996). The CATHENA gives the initial conditions for ELESTRES from which inlet feeder breaks are initiated.

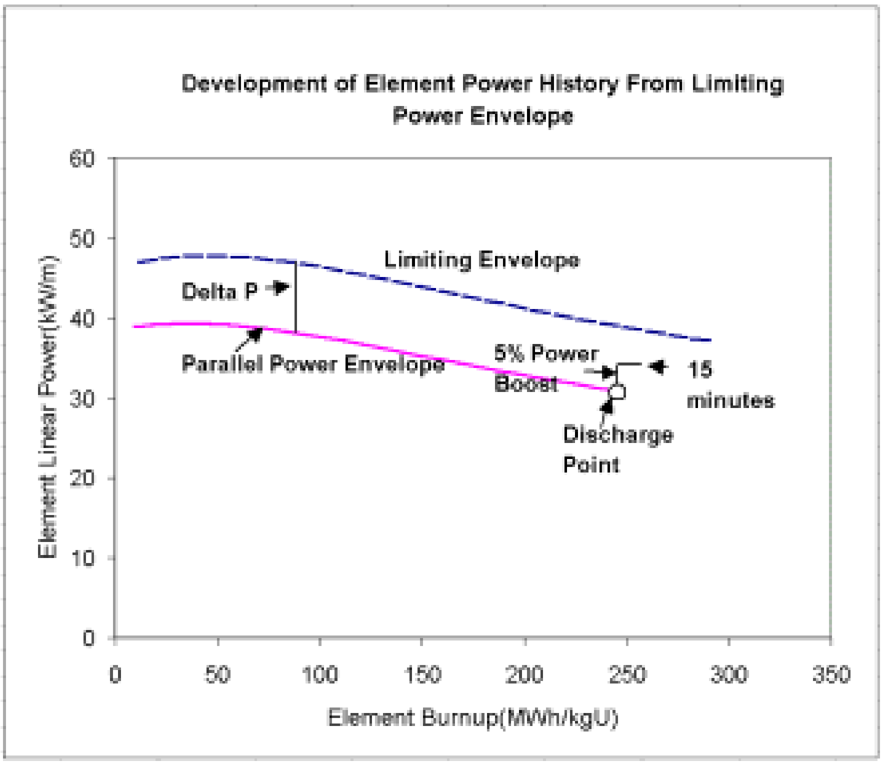

Figure 2. Development of Element Power History from Limiting Power Envelope

4.1.2 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 calculation of transient fission product release from the fuel grains and grain boundary is performed with Gehl's release model (Gehl, 1981). 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 $\left(\frac{dT_{c/1}}{dt}\right)$ in K/s as follows:

$$Zc = 7.58 \times 10^{-19}\ T_{c/1}^{5.7} \left(\frac{dT_{c/1}}{dt}\right)^{-0.346} \qquad (1)$$

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.

Table 5
Release Criteria Applied in Addition to Gehl's Model to Account for Interaction Between Zircaloy and Uranium Dioxide

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

4.1.3 Release Calculations – Off- Stagnation Breaks

The fuel conditions at the time of the accident which are estimated using the ELESTRES code are used as part of the input to the transient analysis code ELOCA·Mk5m (Walker et al, 1992). ELOCA·Mk5m is used to simulate a single fuel element primarily for the transient thermo-mechanical response following the accident. In addition to the fuel initial conditions supplied via ELESTRES, the ELOCA·Mk5m code uses the transient coolant conditions from the CATHENA simulations and a constant relative power. For the off-stagnation break scenarios, additional release of fission gas to the gap is expected to be significant because of the relatively extended period of operation at higher than normal temperatures. For some elements, this increased gas pressure can cause the sheath to strain and lift off the fuel pellets. This enhances the potential for sheath failure due to over-strain. Increased internal gas pressure also has the effect of reducing the heat transfer from fuel to sheath, thereby increasing fuel temperatures. Release of gas during the transient is a strong function of pellet temperature. The time variation of the volume of fission gas released is defined by Tayal (Tayal et al, 1983) for a given volume-average fuel temperature as:

$$V(t) = \frac{a}{\frac{b}{t}+1} \qquad (2)$$

where t is the time and a and b are constants. The constants a and b are determined using the data in Table 6.

Table 6
Additional Gas Release for Hold Times of 200, 1000 and 2000 seconds

Volume Average Fuel	Additional Fission Gas Release Volume in mm^3

Temperature (°C)	after 200 seconds	after 1000 seconds	after 2000 seconds
2069	22370	35298	N/A*
1862	9148	21016	N/A*
1681	3213	9930	14218
1657	2653	8794	N/A*
1599	N/A*	5900	9154

* N/A = data not available

Table 6 gives the additional fission gas volume released when fuel is held in dryout for a period of time at a constant volume-average temperature. Two boundary conditions are required to calculate the constants in Equation 2. For a given temperature, the constants a and b are estimated by specifying the volume released at two different hold times from Table 6. The values of the constants for volume average fuel temperatures which are not in Table 6 may be determined by interpolating or extrapolating to obtain the volume release at two hold times. The ELOCA·Mk5m is used to determine the behaviour of the fuel and fuel sheaths in each bundle, during the transient. These simulations provided information on the fuel temperature transients as well as timing of fuel failures during the accident. This information was used to predict timing of the start of UO_2 oxidation by steam and to estimate the UO_2 oxidation rates. In addition, these results are used to estimate the extent of Zircaloy/UO_2 reactions during the transient. Fuel sheath behaviour has been assessed to determine the number of fuel elements that are predicted to fail during the transient and to determine the timing of these failures. This assessment of the sheath behaviour was performed by comparing the ELOCA·Mk5m simulation results against the sheath failure criteria. Sheath failures were assumed to occur if the ELOCA·Mk5m results indicated that any of the following failure criteria were met or exceeded:

a) 2% sheath hoop strain and sheath temperatures greater than 1000 ^{o}C.
b) 5% sheath hoop strain at any sheath temperature.
c) Fuel centreline melting (T_{CL} greater than 2840°C).
d) Oxygen concentration in the sheath greater than 0.7 weight% over at least half of the cladding thickness.
e) Probability of beryllium-braze assisted cracking greater than 1%.

Following sheath failure, fission products continue to be released from the fuel. These additional releases arise from continued operation at elevated temperature and due to enhanced diffusion cause by steam oxidation of the UO_2. Oxidation of UO_2 pellets leads to a direct enhancement of the diffusional release of fission products from the fuel matrix. The model used to determine the increase in the diffusion coefficient is the one proposed by Lewis et al (Lewis et al, 1990), which in turn is based on the analysis of Turnbull (Killeen et al, 1987). In Turnbull's treatment the diffusion coefficient is represented as a composite expression including a term representing the uranium vacancy concentration as a function of the deviation from stoichiometry (x). Turnbull's diffusion coefficient consists of three terms, each of which represents the behaviour in different ranges of temperature.

$$D(x,T) = 7.6 \times 10^{-10} \exp\left(\frac{-70000}{RT}\right) + S^2_{jv}(V + V_u) + 2 \times 10^{-40} \qquad (3)$$

where,
D = effective diffusion coefficient (m^2/sec)

x = deviation from stoichiometry
T = temperature (K)
R = gas constant (Cal/mol K)
j_v = vacancy jump rate
= $10^{13} \exp(-5.52 \times 10^4/RT)\ sec^{-1}$
V = irradiation-induced vacancy concentration
V_u = uranium vacancy concentration
= fission rate (fissions/m^3 sec)

and

$$V = (9 \times 10^{-5} + 100V_u)/200\ \{[1 + 0.08/j_v\ (9 \times 10^{-5} + 100V_u)^2]^{1/2} - 1\} \quad (4)$$

$$V_u = Sx^2/F_0^2\ \{\ [^1/_2 + F_0/x^2 + 0.5\ (1 + 4\ F_0/x^2)^{1/2}\ ] \quad (5)$$

where,
S = exp (-147200/RT)
F_0 = exp (-71300/RT)

The first term in Equation 3 accounts for intrinsic diffusion at high temperature; the second irradiation-enhanced vacancy production at intermediate temperatures; and the third, irradiation-enhanced athermal diffusion at low temperatures (Killeen et al, 1987). The second term incorporates the effect of increasing stoichiometry. A two-fold increase in the order of magnitude of the diffusion coefficient may be expected when 'x' increases from 0 to 0.1 at constant temperature. Using this expression, it is possible to determine the increase in diffusion coefficient which results from oxidation from UO_2 to UO_{2+x}. If the Turnbull diffusion coefficient is designated D(x,T), where 'x' is the deviation from stoichiometry and T is the temperature, then the increase in the diffusion coefficient due to oxidation is:

$$H(x) = \frac{D(x,T)}{D(x=0,T)} \quad (6)$$

Lewis' methodology consists of applying this correction factor to the empirical diffusion coefficient in the ANS 5.4 model (ANSM, 1982) as employed in the ELESTRES code.

$$D\ (x,T) = D(T)\ H(x) \quad (7)$$

where,

$$D(T) = \left[3 \times 7.8 \times 10^{-10}\ \exp\left(\frac{-Q}{RT}\right)\right]$$

Q = 287 kJ/mole

Once the modified diffusion factor is known, the releases from the fuel are determined using a simple Booth diffusion model (Notley and Hastings, 1980). The Booth diffusion model gives the cumulative release fraction over a time period 't' as:

$$f = 6\ \sqrt{\frac{Dt}{\pi a^2}} - \frac{3Dt}{a^2} \quad (8)$$

where D is the diffusion coefficient and a is the radius of a diffusion volume which is assumed to be spherical. In this case, 'a' is the grain radius. For conservatism, 'a' may be assumed to be equal to the grain radius of the unirradiated fuel as used in the ELESTRES simulations. Equation 8 with the modified diffusion coefficient (Equation 7) is used to calculate the additional releases due to oxidation during various time periods throughout the transient. However, the calculation of the enhancement in the diffusion rate is done at every time step. As a result, the instantaneous value of the diffusion coefficient varies with time due to the changes in fuel oxidation and due to changes in the fuel temperature. However, the fission product releases are estimated (using Equation 8) over the length of the transient. Therefore, a running value of the time-averaged diffusion coefficient is used to estimate the fraction of the grain inventory released due to oxidation. These releases are superimposed on the additional gas releases due to high fuel temperatures.

5. Analysis Results

5.1 Feeder Stagnation Break Results

The Stagnation break area was taken to 17.75 cm^2 from thermalhydraulic analysis. 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 7 with that of 37 standard element from the Wolsong NPP Safety Report. The release calculations were based on CATHENA fuel temperature predictions. The total cumulative fission product release of CANFLEX-NU is quite lower than that of 37 standard element analysis (96922 TBq, 34% of the total inventory).

Table 7
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
Xe-133	4405.3	6538.5
Total isotope	67285.5	96922.2

5.2 Feeder Off-stagnation Break Results

The 37 cm^2 break area results are taken to represent the limiting off-stagnation break. ELOCA·Mk5m simulations have been performed to determine the behaviour of the fuel and fuel sheaths in each bundle, during the transient. These simulations provided information on the fuel temperature transients as well as timing of fuel failures during the accident. This information was used to predict timing of the start of UO_2 oxidation by steam and to estimate the UO_2 oxidation rates. The results of the ELOCA·Mk5m simulations also required thermalhydraulic boundary conditions (coolant temperature, channel pressure and the sheath to coolant heat transfer coefficient). These thermalhydraulic boundary conditions were obtained from CATHENA SLAVE simulations for channel O6_mod. The thermalhydraulic boundary condition is a strong function of the axial location within the fuel channel. Figure 3 shows the volume averaged fuel temperatures for the outer elements of bundles 5 through 9. The volume averaged temperature increased throughout the transient because fuel centreline temperature did not reach the fuel melting point (3113°K), which makes UO_2 expand and contact the fuel sheath (shown in Figure 4).

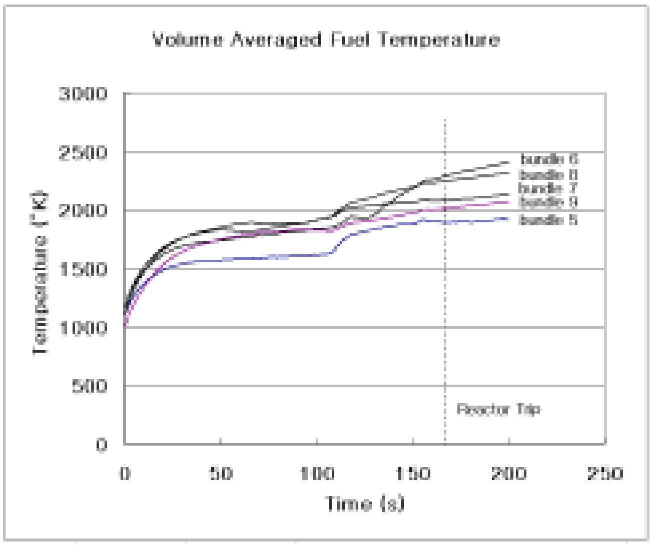

Figure 3. Volume Averaged Fuel Temperature Transients for the Outer Elements of Bundles 5 Through 9 during the Off-Stagnation Feeder Break

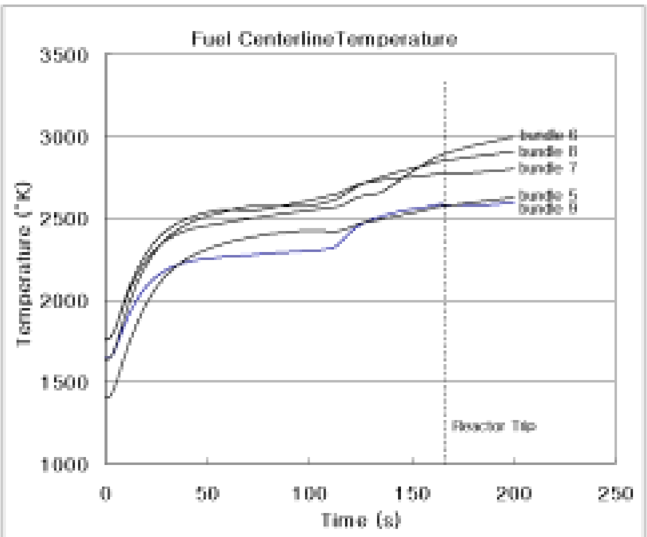

Figure 4. Fuel Centerline Temperature Transients for the Outer Elements of Bundles 5 Through 9 during the Off-Stagnation Feeder Break

Figure 5 shows the fuel sheath temperature transients for the outer elements of bundles 5 through 9. The highest sheath temperatures are found in the outer element of bundle 8. Analysis of the simulation results indicates that the majority of the sheath temperatures remain above 1000°C (1273°K) throughout the transient. Only the outer elements of bundle 5 have sheath temperatures below 1000°C prior to reactor trip.

Analysis of the ELOCA·Mk5m results indicated that only the outer elements of bundles 6 and 8 were predicted to the sheath fail. In all two cases, the sheath failures were caused by more than 2% sheath hoop strain and sheath temperatures greater than 1000 °C, one of the sheath failure criteria discussed above. The earliest sheath failure occurred in bundle 8, 146 seconds after the accident. The failure time for bundle 6 occurred after 151 s.

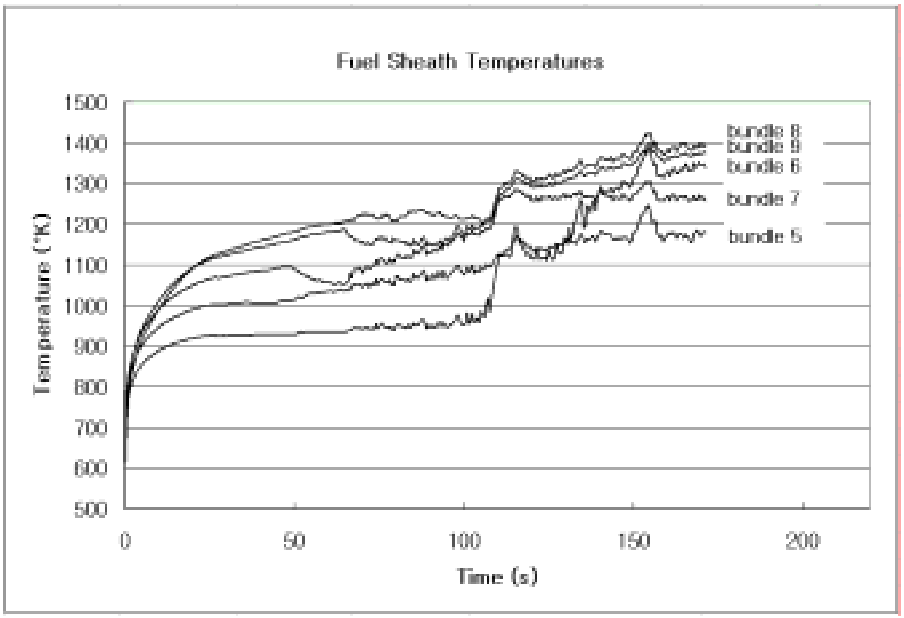

Figure 5. Fuel Sheath Temperature Transients for the Outer Elements of Bundles 5 Through 9

The results of these ELOCA simulations indicate that the assumption of all fuel elements fail after 146 seconds is a significant under prediction of the time required for element failure. As a consequence of this assumption, the releases of fission gas are over-estimated. Furthermore, the extent of UO_2 oxidation is also over-estimated in this analysis. These estimates of UO_2 oxidation are used to estimate the enhancement in fission product releases (due to oxidation) from the fuel.
Table 8 shows the fraction of the grain inventory released due to oxidation-enhanced diffusion at the time of reactor trip (171 seconds).

These release fractions have been used to determine the fraction of the grain inventory that is released from each fuel element. These releases are combined with the gap inventory and grain boundary inventory estimates plus the releases due to operation at elevated temperatures, to determine the total release from each element. The total release from the channel is predicted to be 77064 TBq or 27% of the total inventory. The total fission product release of CANFLEX-NU fuel for off-stagnation feeder break is lower than that of 37 standard element analysis (89925 TBq). The initial fission product release and fuel temperatures during the transient of the CANFLEX-NU fuel in general are lower than 37 element fuel, which leads to lower oxidation releases, and the elements fail later in the transient. These factors contribute to the overall reduction in fission product release from CANFLEX-NU fuel compared with 37 element fuel.

Table 8
Fraction of Grain Inventory Released Due to Steam Oxidation of the Fuel

Bundle	Element	Fraction Released	Bundle	Element	Fraction Released
bundle_1	centre	.009	bundle_3	inner	.026
bundle_1	inner	.009	bundle_3	inter	.028
bundle_1	inter	.009	bundle_3	outer	.032
bundle_1	outer	.013	bundle_4	centre	.041
bundle_2	centre	.019	bundle_4	inner	.054
bundle_2	inner	.020	bundle_4	inter	.031

bundle_2	inter	.023	bundle_4	outer	.059
bundle_2	outer	.026	bundle_5	centre	.099
bundle_3	centre	.025	bundle_5	inner	.129
bundle_5	inter	.044	bundle_9	inner	.069
bundle_5	outer	.136	bundle_9	inter	.045
bundle_6	centre	.146	bundle_9	outer	.078
bundle_6	inner	.169	bundle_10	centre	.052
bundle_6	inter	.062	bundle_10	inner	.053
bundle_6	outer	.189	bundle_10	inter	.057
bundle_7	centre	.146	bundle_10	outer	.063
bundle_7	inner	.169	bundle_11	centre	.009
bundle_7	inter	.062	bundle_11	inner	.009
bundle_7	outer	.189	bundle_11	inter	.009
bundle_8	centre	.100	bundle_11	outer	.012
bundle_8	inner	.131	bundle_12	centre	.009
bundle_8	inter	.041	bundle_12	inner	.009
bundle_8	outer	.137	bundle_12	inter	.009
bundle_9	centre	.055	bundle_12	outer	.010

6. Conclusions

- The fission product release assessment is performed for the stagnation and off-stagnation inlet feeder breaks. For the stagnation break event, the break area was 17.75 cm^2 and 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).
- For the off-stagnation break event, the 37 cm^2 break area results are taken to represent the limiting off-stagnation break and total release from the channel is predicted to be 77064 TBq, 27% of the total inventory. The fission product release of CANFLEX-NU fuel is lower than that of 37 standard element analysis (89925 TBq). The initial fission product release and fuel temperatures during the transient of the CANFLEX-NU fuel in general are lower than 37 element fuel, which leads to lower oxidation releases, and the elements fail later in the transient. These factors contribute to the overall reduction in fission product release from CANFLEX-NU fuel compared with 37 element fuel.
- In terms of dose consequences, those releases of radionuclides to the environment are limited such that public doses are below the acceptable limits.

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