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FUEL BEHAVIOUR DURING LARGE BREAKS IN THE PRIMARY HEAT TRANSPORT CIRCUIT

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ABSTRACT

A large break in the Primary Heat Transport System (PHTS) is considered each break with a size greater than the largest feeder diameter. The break discharges coolant to the containment, causing PHTS depressurization and consequent increase of temperature and pressure inside containment. The PHTS depressurization induces coolant voiding and, due to the positive reactivity void coefficient, power increases until reactor shuts down on a neutronic or a process-conditioned trip parameter. During the power pulse phase, due to degraded fuel cooling, the sheath may fail.

The heat transport system flow decreases faster in the core pass downstream the break. Some channels in the broken loop may become steam filled and others can experience stratified two-phase flow, exposing some fuel elements to steam cooling, inducing fuel temperature rises. A rise in fuel temperature increases the internal fuel element gas pressure, whereas a rise in sheath temperature reduces the sheath strength. If the channel coolant pressure falls below the fuel element internal gas pressure, the sheath stress is increased. Increased internal fuel element gas pressure, along with the decreased coolant pressure, increases fuel sheath stresses. If fuel temperature becomes high enough, sheath failure can occur in a large number of fuel bundles, releasing fission products to the coolant.

One of the challenges met during the fuel analysis was to set a credible, yet conservative "image" of the in-core fuel power/burnup distribution. Consequently, a statistical analysis was performed to find the best-estimate plus uncertainties map for the power/burnup distribution of all in-core fuel elements. For each power/burnup bin in the map, the fission product inventory and the fuel parameters at the end of the steady-state irradiation stage were computed. Afterwards, for each power/burnup bin in the map, the fuel behavior is simulated during the transient. Based on the fuel failure criteria, the failed fuel elements are identified, providing the total radioactive release to the coolant circuit, base for the final radioactive releases outside containment and dose assessment.

The present paper reviews the methodology and results for typical Large Loss Of Cooling Accident with All Safety System Available. Methodologies used in the analysis and results, focused upon fuel behavior, are presented.

Introduction

A large break for CANDU type of reactors is defined as a break with a size greater than the diameter of the largest feeder from the Primary Heat Transport System (PHTS). At CANDU 6 reactor, a large break is any break larger than about 2.5% of the reactor inlet header cross-sectional area.

The consequences arising from a postulated large break in the PHTS are examined since such event could lead to a degraded fuel cooling in a large number of fuel channels. The increase in fuel or sheath temperature could lead to fuel failure and, hence, to a consequent release of fission products to the coolant. Analysis of these breaks provides a basis for assessing the design and effectiveness of the safety systems performance. Three break locations are assumed in the Large Loss Of Cooling Accident (LOCA) type of analyses for CANDU reactors: Large break in the Reactor Outlet Header (ROH break); Large break in the Pump Suction Header (PSH break) or Large break in the Reactor Inlet Header (RIH break).

1. Large Break LOCA – Event Sequence

Wherever the postulated break is located, a certain sequence of events develops:

a) A large break occurs in a large diameter pipe of the PHTS, discharging coolant to the containment. Pressure, temperature and humidity of the containment atmosphere increase.
b) PHTS depressurization causes coolant voiding and, consequently, an increase in core reactivity.
c) For large breaks exceeding certain limits, Reactor Regulating System cannot compensate the rate of reactivity increase. Power increases until the reactor shuts down on a neutronic or a process-conditioned trip parameter. The net effect is a short overpower pulse followed by power rundown to fission products decay power.
d) Containment isolation is automatically initiated on a high reactor building pressure signal. The high reactor building pressure signal also conditions the Emergency Core Cooling System (ECCS) injection signal and the steam generator crash cooldown signal.
e) PHTS loses inventory and depressurize at a rate depending on break size and location.
f) PHTS flow decreases faster in the core pass downstream the break. If the break is large enough, the flow will reverse in the pass. For some break sizes, flow momentarily falls very low as the break upstream of the core pass balances the pumps. Some channels may become steam filled and others experience stratified two-phase flow, exposing some fuel elements to steam cooling. Fuel and sheath temperatures rise. A rise in fuel temperature increases the internal fuel element gas pressure, while a rise in the sheath temperature reduces sheath strength. Channel coolant pressure falls below the fuel element internal gas pressure, stressing the sheath. If sheath temperature becomes high enough, sheath failure can occur.
g) Following the reactor trip, fuel temperature decreases as the heat generation rate decreases and the temperature profile in the fuel pin flattens out.

h) ECCS is activated when the broken loop pressure falls below specified setpoint and is followed by the isolation the two PHTS loops.
i) Soon after PHTS loops isolation, ECCS injection refills the broken loop. As a result, fuel and sheath temperatures start to decrease. Depending on their initial temperatures, some fuel sheaths may fail due to thermal shock following rewet.
j) If fuel sheath fails, some fission products are released to the coolant and are carried into containment through PHTS piping break. They can become airborne in the containment atmosphere although most of the soluble radioactive material is carried out with the liquid phase to the reactor building floor. Once in the containment atmosphere, fission products plate out on the walls and internal surfaces. Some are washed out by dousing and some will decay away. During containment over-pressure period, some airborne fission products could leak to the outside environment.
k) Long term cooling of the broken loop is maintained by the ECCS flow through the circuit, with decay heat removed by ECCS heat exchangers and through the break while the intact loop cooling is maintained by thermosyphoning, with decay heat removal by steam generators.

2. Trip Coverage

The design trip parameters for Large LOCA events are the high neutron power and the high neutron power log rate. Other relevant trip parameters for these events could be high reactor building pressure, low PHTS pressure or low PHTS differential pressure.

3. Methodology for Reactor Physics Analysis

The power transient analyses are simulated by coupling the thermalhydraulic code CATHENA with the reactor physics code RFSP.

3.1 Heat transport circuit model for thermalhydraulic simulations (CATHENA code)

The thermalhydraulic part of the analysis is performed by using a *two-loop*, *multiple average channel* circuit model. Since power pulse depends mainly on voiding rate within channels downstream the break, these channels are grouped into seven average channels, based on channel power, channel elevation and header/feeder connection elevation.

Analyses are performed for an aged plant, at 5500 FPD (about 18 years of service at 85% FP). Due to increase in bundle power and reduction in coolant flow and pressure, during the initial blowdown phase of the accident, many fuel elements will dry-out during the overpower transient. If, from thermalhydraulic point of view, the onset of dry-out would maximize fuel and sheath temperatures, from the neutronic point of view, dry-out reduces vapor generation (coolant voiding) and, in turn, the resulted power pulse decreases. Thus, in a conservative manner, during blowdown phase, complete dry-out was assumed in the thermalhydraulic analysis.

3.2 Core model for neutronic calculations (RFSP code)

For the reactor core at equilibrium, the starting point for a Large LOCA transient is achieved in several steps. From a time-average calculation, an equivalent instantaneous core configuration is obtained (reproducing the time-average power distribution, burnup distribution and core reactivity). Then, a long shutdown, followed by a restart to near full power is simulated. The long shutdown simulation is needed to maximize the poison concentration within moderator (~ 2.3 ppm of boron), as the presence of poison increases void reactivity effect during LOCA. The reactor power level at the moment of LOCA initiation was determined by imposing a side-to-side power tilt of about 8%. The break is assumed to occur on the high-tilt side of the core, conservatively enhancing void reactivity and power pulse.

4. Methodology for Thermalhydraulic Analysis (Circuit and Single Channel)

The thermalhydraulic response of the plant during the imposed transient is simulated using the thermalhydraulic code CATHENA. The heat transport circuit model consists of two loops and includes the PHTS, the steam and feedwater systems, part of the reheater drains system, part of the blowdown system and the ECCS. Seven average channels in parallel are represented on each of the four core cooling passes.

For the Single Channel Analyses, a number of six channels are modeled in CATHENA: A9, B10, O6, O6-mod, S10 and W10. The selection is based on channel elevation, channel power and header/feeder connection elevation. Note that channel O6-mod has the same geometry as channel O6 but the channel power and powers on the two central bundles are modified to the licensing limits of 7.3 MW/channel and 935 kW/bundle, respectively. Each model includes the feeders, end fitting and fuel channel. The inlet and outlet header conditions, imposed as boundary conditions, are taken from circuit simulations.

For equilibrium core, a number of three transients were analyzed:

- 100% ROH break,
- 35% RIH break,
- 50% PSH break.

Mentioned locations and sizes for the break are selected from a break survey, with maximum time in stagnant conditions and maximum sheath temperature as selection criteria.

5. Methodology for Fuel Behavior Analysis

5.1 Analysis steps

Since intact loop is expected to be well cooled, the objective of the fuel analysis will be to estimate the quantity and timing of fission products release from failed fuel to coolant in the broken loop. The interaction between codes is presented in Figure 1.

Activation of shutdown systems and ECCS ensures that the period of fuel heat-up will be short and the extent of fuel failures will be, likewise, limited. If sheaths fail, fission products from those fuel elements are available for release. However, sheath failure does not imply immediate release of fission products from grain surface or from within grains. Releases from grain to gap are temperature and time dependent. For large breaks, where fuel heat-up period is not long, release of fission products from grain surface or from grain boundary is expected to be less than 1% of the total inventory within pellet. Moreover, free gap inventory is not released immediately following sheath failure. Examinations of fuel elements with high gas release, operated at high powers, have shown deposits of some fission products on the sheath inside surface. Iodine is expected to chemically combine with Cs and be retained on fuel and sheath surfaces. Noble gases, such as Kr and Xe, are expected to be released mostly at the time of sheath failure, since they are not chemically active.

Calculation of fission products release is done by determining the number of failed fuel elements and the timing of failure. Conservatively, it is assumed that radioactive release from failed fuel element consists of the total gap inventory plus 1% of grain inventory. Also, it is assumed, in the same conservatively manner, that release occurs at the time of failure.

Basic steps of a fuel failure analysis are the following:

1. A preliminary Sensitive Analysis on fuel design parameters is necessary for evaluating their impact on gaseous fission products fractional release from fuel matrix to the gap. Fuel design parameters are modified within a $\pm 2\sigma$ range and the combination that maximizes the fractional release is selected to be further used in fuel behavior simulations with ELESTRES.
2. Obtain power/burnup distribution of the in-core fuel elements. This distribution must be a consistent “image” of the reactor core during steady operation at full power.
3. Derive a credible, yet conservative, set of irradiation histories (linear power vs. burnup) for all in-core fuel elements that should cover “real” histories under normal operating conditions. The irradiation histories will be produced based on the so-called Limiting Overpower Envelope (LOE).
4. Evaluate fission products inventory (total and within gap) for fuel elements grouped in each bin of the power/burnup distribution.
5. Predict power/burnup distribution of the fuel elements expected to fail, based on the Fuel Failure Threshold methodology.

5.2 Sensitive analysis on fuel design parameters

The result of the sensitive analysis on fuel design parameters is given in Table 1.

5.3 Power/Burnup distribution of in-core fuel elements

During reactor operating history, fuel elements experience a wide spectrum of power/burnup values, while irradiation changes continuously. If an accident analysis is to be performed then a period reasonably long of the reactor lifetime must be analyzed. It means that several millions of simulations are required to study fuel behavior for all in-core fuel elements – obviously, an unreasonable time consuming process. To avoid it, the alternative is to derive, by a statistical analysis, the most "representative", yet conservative, power/burnup distribution of the fuel elements within reactor core. An analysis extended over two years of operation gives a consistent "image" of the fuel elements distribution within an equilibrium core, at full power, valid at any moment of the reactor lifetime.

Reactor operation is routinely tracked, every few days of operation at high power, by core simulations performed with the RFSP code. The results obtained from each simulation are used to plot the number of fuel elements in each bin of a linear power vs. burnup map. The ranges for fuel element linear power and burnup were selected to cover all possible values during reactor operation at full power: 1 – 65 kW/m, in steps of 1 kW/m, for linear power and 10 – 270 MWh/kgU, in steps of 10 MWh/kgU, for burnup. A statistical analysis was performed on the set of power/burnup maps (one map for each core simulation). Finally, the power/burnup Best Estimate Distribution (BED) of the fuel elements within the reactor core was obtained by plotting the average number of fuel elements in each power/burnup bin. The corresponding standard errors were also calculated to be used in obtaining the best estimate plus uncertainty map – the Limit Estimate Distribution (LED). Figure 2 gives the LED map, with a 95% level of confidence.

To account for the errors in power calculation with RFSP, the fuel elements powers, obtained from core simulations, are, conservatively, increased by 3%, producing the map for 103% FP, to be used in further fuel behavior analyses (Figure 2).

5.4 Limiting overpower envelope (LOE)

Both thermo-mechanical behavior and radioactive nuclide inventory of a fuel element under normal operating conditions are predicted by the ELESTRES computer code and depend on irradiation history (linear power vs. burnup). Since the number of ELESTRES simulations necessary to cover all possible irradiation histories is unreasonable high, the alternative is to derive a limited set of power/burnup histories, consistent with the real ones occurring in-core.

The curve plotting maximum fuel element linear power reached in each burnup bin in the LED map is the so-called Reference Overpower Envelope (ROE). Since this curve is derived from a limited number of core simulations, it is possible for some fuel elements to exceed it, during the reactor lifetime, for a short time, due to unusual or abnormal fuelling or due to short-term power control transients. However, throughout reactor lifetime, most of the in-core fuel elements are expected to have their irradiation histories covered by ROE.

Starting from ROE, the so-called Limiting Overpower Envelope (LOE) is produced by scaling ROE in the way its peak corresponds to the linear power on an element from the outer ring of a bundle, operating at the license limit of 935 kW/bundle.

It is assumed that the irradiation history for any in-core fuel element, at any time in the operating history, follows LOE shape. Thus, scaling LOE to the current linear power and burnup, the irradiation history can be derived for all in-core fuel elements (Figure 3).

5.5 Fuel failure threshold

For a given burnup, fuel failure threshold is given by the maximum linear power for which a fuel element, operating under the accident transient conditions, is predicted not to fail (Figure 2). Fuel element behavior during accident is simulated by the ELOCA computer code using as input data:

- pre-transient fuel element thermo-mechanical data supplied by ELESTRES simulations (note that these simulations provide also the fission products inventories within gap and pellet),
- transient thermalhydraulic boundary conditions (coolant temperature, coolant pressure and heat transfer coefficient from sheath to coolant) predicted by a Single Channel Analysis (using CATHENA code). Conservatively, it is assumed that all in-core fuel elements will experience the conditions from top, outer ring, fuel element of bundle 7, in the O6-mod fuel channel.

Among the results of the transient simulations, the ELOCA code calculates: fuel temperature, sheath temperature, sheath strains, pressure within gap, sheath oxidation level. Based on ELOCA simulations results, simple and conservative criteria are used to determine whether a fuel element fails or not. They are based on experimental data and experience in reactor operation. Fuel sheath is considered to remain intact if the followings are satisfied:

a) *No fuel centerline melting***.** A fuel element is assumed to fail if fuel centerline melting is reached. Failure occurs due to volume expansion, causing excessive sheath strain.
b) *No excessive diametral strain.* Uniform sheath strain shall remain less than 5% for sheath temperatures lower than 1000 ºC.
c) *No significant cracks in the oxide surface*. Uniform strain shall remain below 2% for sheath temperatures higher than 1000 ºC.
d) *No oxygen embrittlement*. Oxygen concentration shall remain less than 0.7 w% over half of the sheath thickness.
e) *No sheath failure by beryllium-braze penetration* at bearing pad and spacer pad locations.

5.6 Transient fission product release

This part of the analysis uses the results of the fuel failure threshold calculation. Release is estimated by summing the contribution from all fuel elements predicted to fail (above the failure threshold curve).

6. Results

Figure 4a and 4b show some examples for fuel sheath behavior during the 100% ROH break transient for two different burn-up values. The first case (Figure 4a) corresponds to a low relatively low burnup. Even the fuel sheath temperature increased significantly during the transient, fuel sheath failure did not occur until the strain increased above the 2% limit (significant cracks in the oxide surface). For the second case (Figure 5b), a larger burnup case was considered. For this case, the fuel sheath strain is relatively high but fuel sheath failure occurred only when the fuel sheath temperature increased over the 1000^0C limit.

Figure 5a and 5b show some examples for fuel sheath behavior during the 35%RIH break transient for two different burn-up values, similar to the previous case. In the first case (Figure 5a), even the fuel sheath temperature increased significantly, fuel failure did not occur because the fuel sheath strain remained low over the analyzed period. For the second case (Figure 5b), even the fuel sheath strain increased significantly from the beginning of the transient, fuel sheath failure occurred only when the fuel sheath temperature increased over the 1000^0C limit (significant cracks in the oxide surface).

Similar methodology has been used to evaluate fuel behavior in case of 50% PSH break.

The entire distribution of fuel pins burnups and linear powers was analyzed and, for each case, the time when the fuel sheath failure occurred has been recorded. Based on the specific conditions for that fuel pin, radioactive release from failed fuel element consists of the total gap inventory plus 1% of grain inventory. Conservatively, it was assumed that releases occur at the time of fuel sheath failure. Figure 6 presents the time evolution of the I-131 releases during each transient. The maximum release are obtained for the case of 100% ROH break. Fission product releases has been used for detailed containment analyses and dose calculations.

7. Conclusion

The paper presented a review of the methodology used to analyze a Large LOCA event for Candu 6 reactor with all safety systems available. Typical results for critical breaks were presented from the perspective of the fuel behavior and fission product releases.

Table 1 – Results of Sensitive Analysis on fuel design parameters

Parameter	Max. Temp.	Max. Strain	Max. Inventory
Pellet Diameter	-	-	MAXIMUM
Dish Depth	-	minimum	minimum
Land Width	-	MAXIMUM	MAXIMUM
Pellet Density	minimum	MAXIMUM	minimum
Pellet Roughness	MAXIMUM	minimum	MAXIMUM
UO2 Grain Size	-	minimum	minimum
Pellet Stack Length	minimum	MAXIMUM	MAXIMUM
Axial Clearance	-	minimum	minimum
Radial Clearance	-	minimum	minimum
Sheath Wall Thickness	MAXIMUM	minimum	MAXIMUM
He Fraction in the filling gas	minimum	minimum	minimum
Sheath Roughness	MAXIMUM	MAXIMUM	MAXIMUM

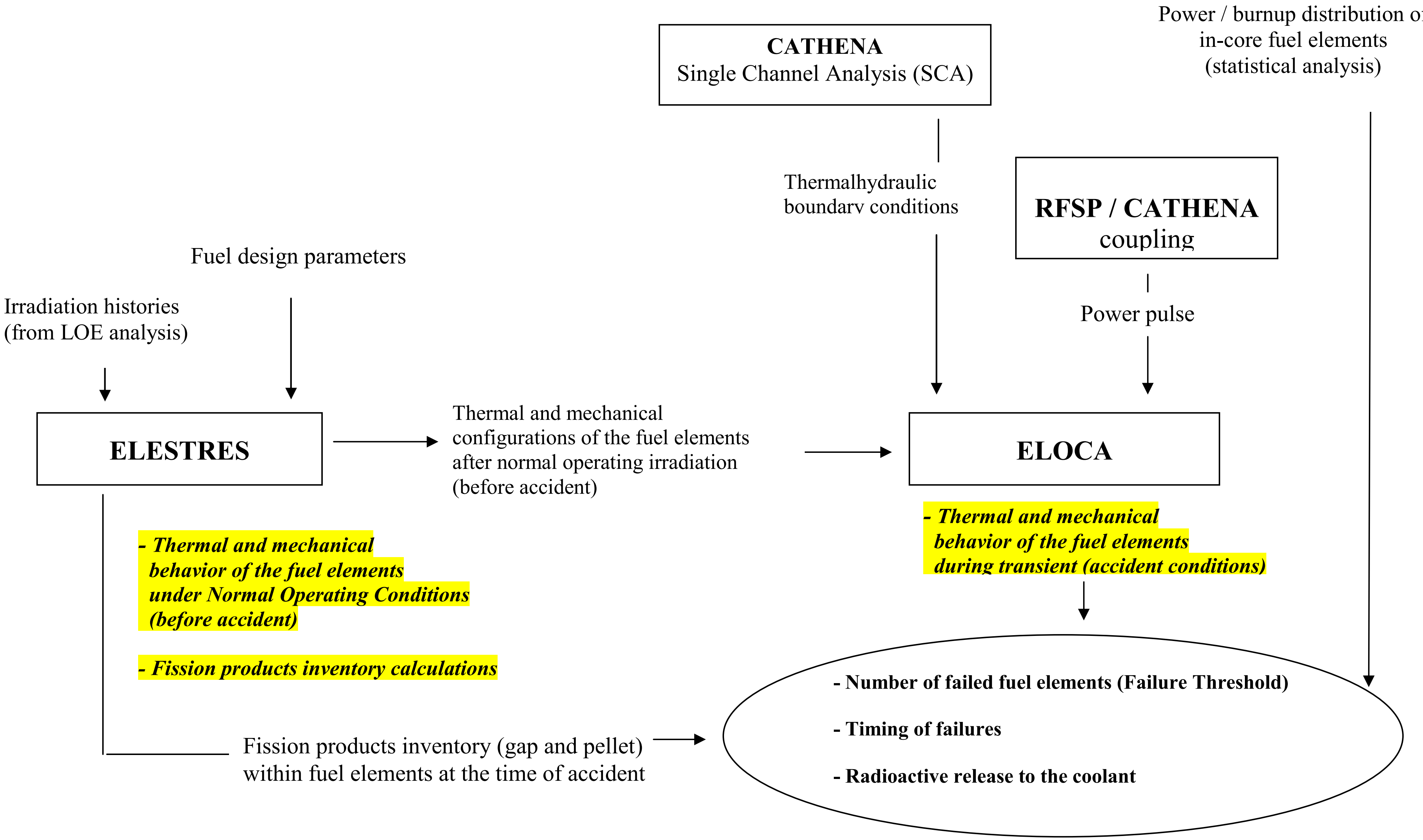

Figure 1 – Diagram for Fuel Behavior Analysis

Linear Power [kW]	Burnup [MWh/kgU]																
	10	20	30	40	50	60	70	80	90	100	110	120	130	140	150	160	170
60																	
59																	
58																	
57																	
56	1		1	2	4	2	1	1									
55	3	4	6	10	10	10	7	3	4		1						
54	11	15	26	27	35	26	20	12	8	5	2	1					
53	30	39	55	63	60	52	48	38	24	16	7	4	2				
52	52	67	61	72	71	75	74	66	57	39	26	16	5	3	2	1	
51	58	78	70	93	80	87	79	77	73	64	55	42	29	11	8	4	
50	93	106	116	123	126	103	110	95	87	81	81	64	60	40	27	18	9
49	118	141	142	129	138	141	139	124	132	108	95	92	74	72	58	45	22
48	118	139	140	138	140	146	145	147	135	139	134	113	95	90	73	65	54
47	131	139	155	149	156	162	155	146	152	159	160	151	135	121	99	89	63
46	150	149	150	145	169	199	200	206	193	168	160	155	161	150	134	112	82
45	145	130	130	133	150	160	170	185	192	207	201	178	166	164	151	144	109
44	125	131	120	120	133	132	148	148	165	187	197	209	197	187	163	140	106
43	114	139	129	138	137	139	135	128	136	150	165	183	212	218	209	188	124
42	120	132	133	144	165	167	151	144	138	133	128	144	169	196	211	195	165
41	150	171	178	192	182	197	191	177	175	148	147	137	141	159	172	165	141
40	206	227	234	251	258	241	228	232	216	202	201	181	157	152	128	128	124
39	220	219	231	237	238	241	249	248	251	259	252	254	228	180	132	100	95
38	229	260	245	258	281	280	263	264	256	271	281	283	287	256	177	131	93
37	230	252	234	248	296	298	297	300	303	289	288	306	290	258	193	148	120
36	215	241	207	244	252	271	275	289	311	329	335	328	300	257	182	126	119
35	197	227	224	236	247	252	267	267	285	307	326	338	326	255	183	131	105

35% RIH Break

50% PSH Break

100% ROH Break

Figure 2 – Limit Estimate Distribution for in-core fuel elements and Failure Thresholds

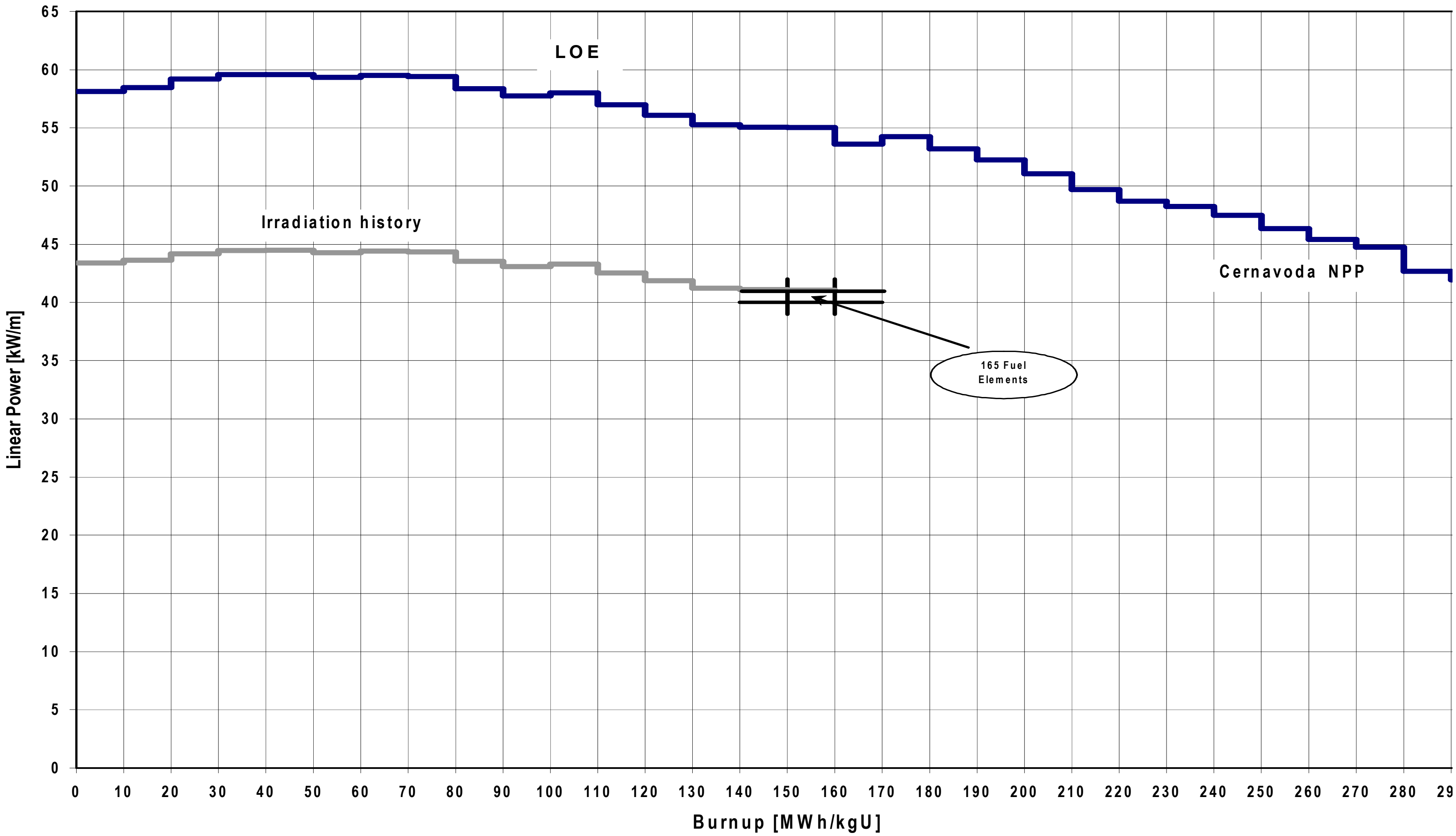

Figure 3 – Limiting Overpower Envelope

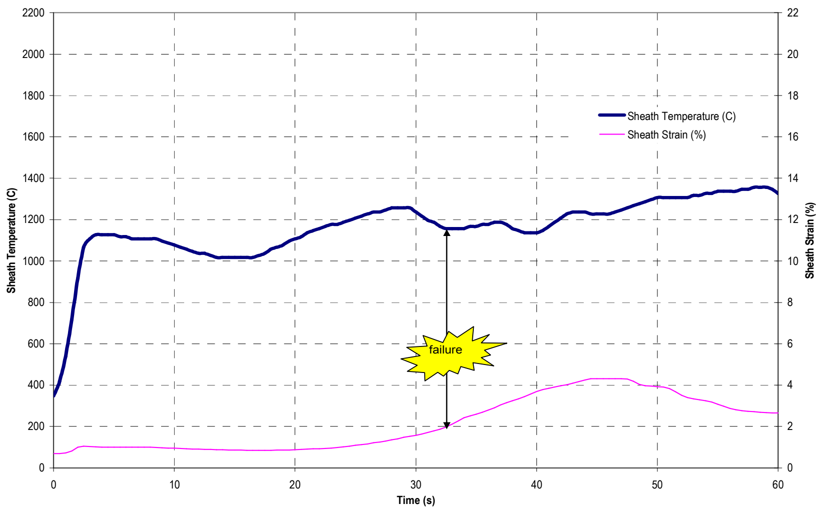

Figure 4a – Sheath temperature and strain for 100% ROH break (burnup = 60 MWh/kgU)

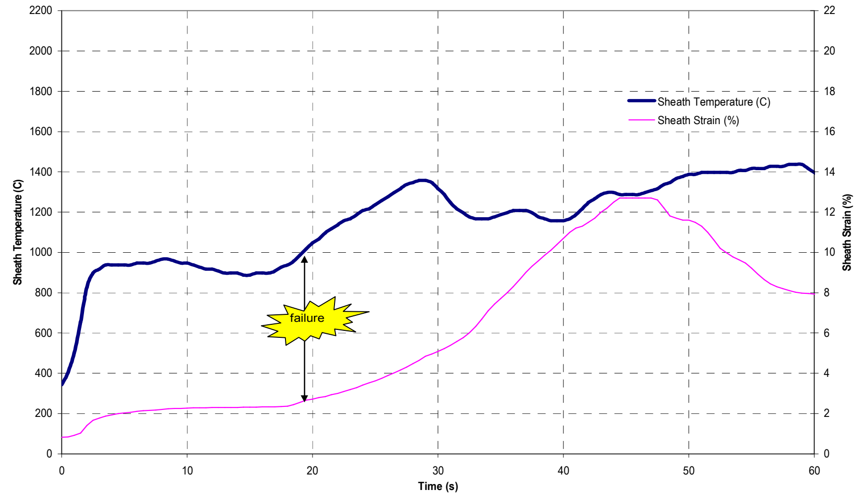

Figure 4b – Sheath temperature and strain for 100% ROH break (burnup = 140 MWh/kgU)

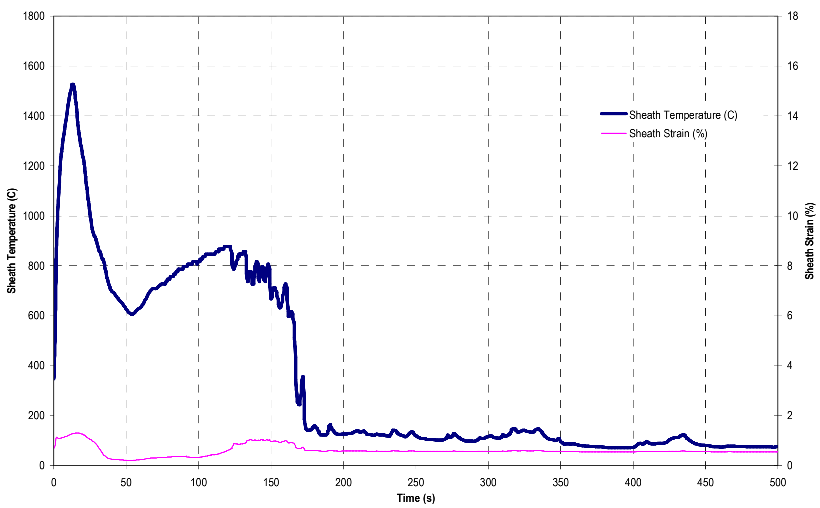

Figure 5a – Sheath temperature and strain for 35% RIH break (burnup = 60 MWh/kgU)

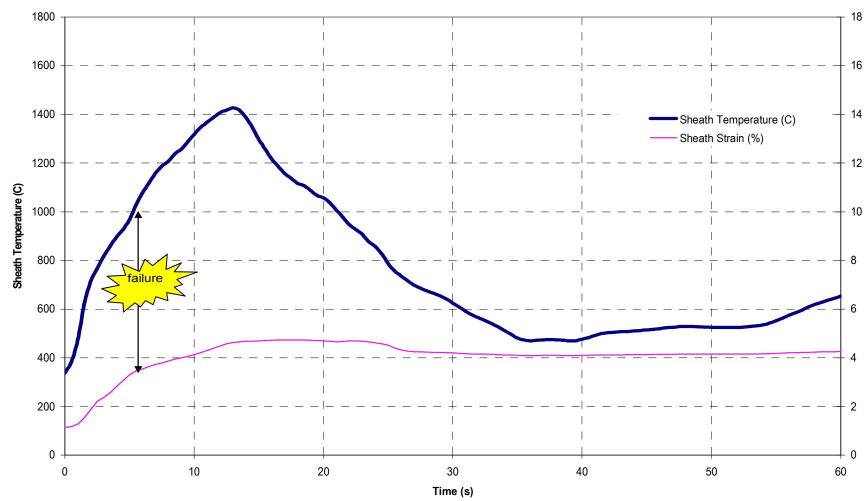

Figure 5b – Sheath temperature and strain for 35% RIH break (burnup = 270 MWh/kgU)

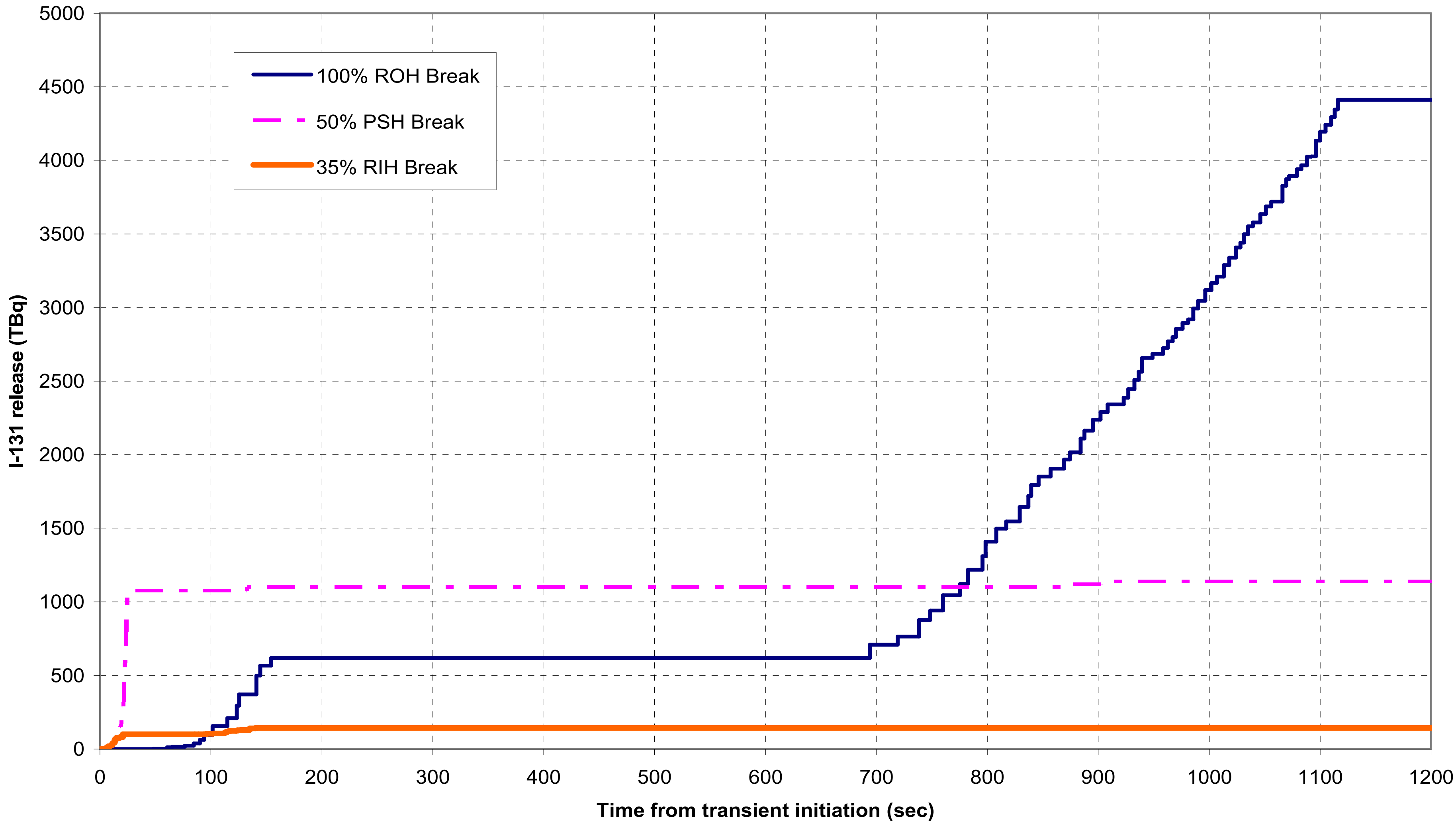

Figure 6 – I-131 inventory release during different accident scenarios