Safety Analysis to Support a Safe Operating Envelope for Fuel

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ABSTRACT

This paper presents an approach for defining a safe operating envelope for fuel. "Safe operating envelope" is defined as an envelope of fuel parameters defined for application in safety analysis that can be related to, or used to define, the acceptable range of fuel conditions due to operational transients or deviations in fuel manufacturing processes. The paper describes the motivation for developing such a methodology. The methodology involved four steps: the update of fission product inventories, the review of sheath failure criteria, a review of input parameters to be used in fuel modelling codes, and the development of an improved fission product release code.

This paper discusses the aspects of fuel sheath failure criteria that pertain to operating or manufacturing conditions and to the evaluation and selection of modelling input data. The other steps are not addressed in this paper since they have been presented elsewhere.

1. Introduction

The mandate of a station safety analysis group is to ensure that the station is operated and maintained in a manner that is consistent with the basis for our understanding of the safety consequences of process failures. As operating experience has accumulated, an awareness of the significance of fuel manufacture, fuel handling, and operating conditions on safety consequences has also grown. This has lead to programs at the station which are designed to ensure that these influences are identified and appropriately considered.

A program of fuel inspections has always been in place at Point Lepreau. But the effort expended in monitoring operating conditions and fuel behaviour has increased. This is the subject of a companion paper in this conference. The influence of these operating conditions on safety analysis and how they could be addressed is the subject of this paper.

A number of Point Lepreau operating incidents have raised the awareness of these issues. The most significant was in 1991 to 1992 in which an above average number of fuel failures occurred. These were ultimately attributed to the underbaking of the Canlub coating during the fabrication process. However, in the same year several hoses in the catenary hose assembly for the fuelling machines were found to exhibit signs of bulging. A white powder-like deposit covered the hoses. These hoses were replaced and the cause was suspected to be an inadvertent over-addition of hydrogen to the heat transport system during a routine hydrogen addition process.

When this incident occurred it was first attributed to higher bundle uranium masses. Fuel manufacturers have refined and improved their production technology and were able to pack more fuel into the same bundle design. So gradually the uranium content of the bundles had been increasing. Because of this suspicion Point Lepreau imposed an upper limit of 19.2 kg U per bundle. In order to accommodate this specification, the manufacturer used a "U_3O_8 add-back" process in which U_3O_8 was blended into UO_2 powder to reduce the pellet densities. Subsequently, in 1995, one lot of fuel powder was produced in which the U_3O_8 was not completely blended in. This resulted in very large pores within the pellet, along with reduced sintered densities. A safety assessment of the implications of this was performed and the fuel was accepted for use.

In order to address issues with heat transport fouling, consideration had been given, in 1994, to treating the heat transport system with a shock of reduced pH. The transient reduction in pH would release magnetite from the inlet side and enable removal of the material by the purification system. A safety assessment of the implications of this magnetite on fuel behaviour was required. As well, as a consequence of leaving a temporary wooden cover in the HTS (see reference [1]), analysis had to be performed. This was to assess the implications of the foreign material, wood in particular, in the heat transport system during the heat soak treatment employed to break the material down and remove it. Some metallic foreign material still remains in the heat transport system due to that event

This accumulation of events increased the awareness that manufacturing variations and plant operating variations can have an influence on the conclusions presented in the Safety Report regarding the safety design effectiveness.

A review was initiated by the Nuclear Safety Department of the safety analysis methods used to predict accident consequences. Four steps were taken:

1. update the fission product inventories and account for the affect of in-core defected elements,
2. review sheath failure mechanisms and assess the implications of incipient defects (or degraded bundle health),
3. review the input parameters used in fuel modeling codes,
4. ensure that a code is provided that is capable of providing fission product releases from fuel under the wide range of accident conditions and that can account for variations in operational states.

2. Fission Product Inventory Update

The work performed here was presented in 1997, see reference [2] and therefore will not be discussed here. The inventory update also provided assessment of the distribution of fission products between grain inventory, grain boundary inventory and gap inventory. The methodology developed provided an approach for accounting for the impact of defected elements in the core on this fission product distribution. Point Lepreau operating procedures to limit the number of defects in-core using heat transport noble gas concentrations, based on the works of references [3] and [4], were credited, although an allowance was made for an operational upset resulting in twice as many defects as would be allowed under these procedures.

3. Sheath Failure Criteria

The implications of normal operating conditions are given here in the context primarily of large break Loss-of-Coolant accidents. During a Large Break LOCA, fuel heatup and HTS depressurization can result in fuel sheath strains sufficient to fail fuel elements, resulting in fission product release from the core. The likelihood of failure of a fuel element is dependent upon many factors. Among these are the initial conditions of the fuel element. The initial conditions of import are fuel and sheath temperature, internal gas pressure, sheath strain, sheath oxidation and sheath hydriding. These conditions will be affected by the fuel element's power/burnup history, the value of various fuel design parameters at the time of manufacture, and the fuel element's environment during operation. Another set of important factors that affect fuel element failure probabilities during a large break LOCA are the transient boundary conditions during the postulated accident. These conditions include the transient power, system pressure and heat removal conditions and force transients experienced by the fuel and sheath due to axial thermal expansion and bundle vibration, acceleration and impact.

The thermal-mechanical behaviour of a single fuel element in specified conditions can be predicted using the ELESTRES/ELOCA combination of fuel element modelling codes. These codes are designed to account for the effect of as-manufactured fuel element design parameters, fuel and fuel sheath properties (ELESTRES only), power/burnup history before the postulated Large Break LOCA and the predicted power transient and the coolant conditions of pressure and heat transfer during the postulated accident. As discussed above, all of these factors affect the fuel element's thermal-mechanical response during the postulated accident and will have a direct impact on the prediction of fuel sheath failures. In summary, the ELESTRES/ELOCA code suite can be used in assessing the impact of the following factors which affect sheath failure:

i. Uncertainties in fuel element design parameters due to manufacturing variations,
ii. Off-nominal reactor operating states such as shim, startups and shutdowns, reduced power operation and fuelling,
iii. Environmental factors during normal operation which could affect fuel sheath properties, such as HTS chemistry and;
iv. Uncertainties in fuel element power, fuel-to-sheath heat transfer coefficients and coolant pressure

transients during the postulated accident

The methodology defined in reference [5] is designed to account for the impact of the first three of these factors on fuel sheath failures during a Large Break LOCA, as the magnitude of the variation of these parameters can be assessed. The impact of the fourth parameter must be assessed in the methodology for the detailed thermal-hydraulic analysis.

In addition to the parameters which can be modelled directly using ELESTRES and ELOCA, there are other processes which can affect the likelihood of sheath failure in a fuel element during a Large Break LOCA. As outlined in the first paragraph of this section, these parameters include:

i. Bundle acceleration and impact due to flow reversal during the blowdown,
ii. Bundle vibration due to blowdown flow turbulence, and;
iii. Axial expansion of the fuel string and subsequent compression due to heatup.

The impact of these parameters on fuel sheath integrity must be assessed in a manner which will allow their impact on sheath failure probabilities to be combined with the ELOCA predictions of sheath behaviour during the Large Break LOCA to yield a combined sheath failure prediction. In order to do this; the additional sheath strains caused by these processes must be estimated. This information can be used as a penalty applied to the sheath failure criteria already built into ELOCA.

3.1 Operating Conditions

The heat transport system operating parameters that can affect the fuel are flows, temperatures, and pressures. These combine to influence the heat transfer coefficient and thus normal operating fuel temperatures and internal element gas pressures. As well, heat transport system pH control, dissolved deuterium and oxygen levels might affect the fuel.

For example, pH Control is largely for the purpose of the purpose of controlling corrosion within the heat transport system. Transients that release crud result in deposition on the fuel sheaths of CANDU fuel. Crud is primarily magnetite which deposits on the fuel sheath by two main mechanisms. The first is chemical deposition/resolution of the magnetite. This is dependent on the solubility of magnetite in the coolant. The second, which is significant only if the solubility of the magnetite is such that deposition/resolution is not significant, is particle deposition due to impaction from flow turbulence.

Crud deposition has never been demonstrated to have an impact on fuel performance (or failure probability), even at deposition concentrations of up to 11 g/m^2, or a thickness of 4.3 μm (Reference [6]). Crud does not increase sheath corrosion rates, so the only impact it might have on fuel performance is through increasing fuel temperatures. Given that magnetite has a thermal conductivity of 0.67 $W/m^2{\cdot}K$ and that the outer element of a 935 kW 37 element bundle has a heat flux of 1.1 MW/m^2, the thermal impact of crud deposition is 1.64 $K/\mu m$. Therefore, the highest observed level of crud deposition would increase the fuel temperature in steady-state operation by 7° K. Crud deposition is therefore limited as a "good practices" measure rather than as a process which can affect fuel failure probabilities.

The most significant crud deposition (11 g/m^2) was observed at Gentilly-1. This unit had very extreme conditions for crud deposition when compared to the CANDU 6 experience. The coolant had ~17 wt% steam when exiting the channel. pH Was controlled with 15-25 mg NH_3/kg D_2O where 50 mg NH_3/kg D_2O gives a pH of 9.7 Therefore, since CANDU 6 reactors have much less boiling than Gentilly-1 and higher heat trransprot pH operating conditions, it is concluded that crud deposition is not a significant factor in fuel sheath failure probabilities.

A second example of an operating condition that could contribute to increased fuel failures in accidents is sheath embrittlement due to deuterium pickup from the heat transport system. Measurements have been performed on 33 elements shipped from Point Lepreau, see references [7] and [8]. The results are illustrated in Figure 1. None of the deuterium concentrations exceeded 190 ppm. This is less than the solubility limit for Zircaloy at temperatures above 400°C. Therefore, in a large break LOCA, hydrogen embrittlement will not be a factor contributing to fuel sheath failures.

Another example of an operational upset would be an inadvertent power manoeuvre that exposed the fuel to overpowers or excessive power ramps. This could result in element defects due to stress corrosion cracking (SCC) or incipient defects due to increased stress levels in sheathing. Normally, this defect mechanism is not observed. However, since the mechanism has an incubation period, incipient defects may exist before detection. The methodology proposed below is thought to be sufficiently robust to address the safety consequences of such an event.

In general, heat transport operating conditions are well monitored and controlled and it was concluded that they are unlikely to contribute to degraded bundle health.

3.2 Safety Analysis Method

The approach taken to assessing the impact of various mechanisms on sheath failure is generally predicated on the fuel being manufactured and operated within the design envelope. However, it is the goal of the sheath failure methodology presented here to explicitly allow for the possibility that a variation outside of the design envelope could occur. The intent is to provide PLGS staff with a way to determine whether the reactor is still operating within the analysed envelope for safety analysis even if a variation occurs. PLGS has programs and practices which ensure that such variations will be detected. When this happens, staff needs to determine, based upon the available information, the number of fuel elements whose resistance to the sheath failure mechanisms described in this document has been significantly degraded.

Our defect detection and management procedures would limit the number of defects in elements having linear power ratings between 50 to 55 kW/m to between about 8 to 12. Assuming that the number of defects detected in an upset condition is 1% of all the elements that have degraded health and assuming that the degradation mechanism affects the higher power elements, not necessarily true but often the case, then this implies that there could be up 1000 elements at risk.

In order to allow for bundles whose resistance to failure is compromised because of degraded bundle health, it is recommended that the sheath failure analysis assume that, after all other failure mechanisms have been accounted for, 250 additional elements will fail at the time of break. The 250 elements should be those 250 which would provide the greatest fission product release of those which do not fail by any mechanism described in this document. The safety analysis should assume that they fail at the start of the scenario. It is proposed that this technique be applied to all accident scenarios for which dose consequence analysis is required.

If this is done, it will be possible to compare the number of elements whose sheath failure mechanisms will be significantly degraded to 250. If the number of bundles in which these elements are distributed is large, then they can be assumed to be distributed evenly throughout the core and therefore only ¼ of them will be affected by a postulated Large Break LOCA. If the number of elements with degraded sheath failure resistance which would be affected by a Large Break LOCA is greater than 250, then the station can be considered to be outside the analysed envelope.

Note that the assumption that 250 elements in addition to those which have already failed implies a certain conservatism, since those elements which are predicted to fail by other mechanisms would have a certain probability of having elements with reduced failure resistance. These elements would likely fail earlier than if their failure resistance had been reduced, but it would mean that the number of *additional* failures would be less than 250.

The assumption of 250 elements with reduced resistance to failure assumes that all of these elements are in the same core pass (the affected core pass for the break). For most possible causes of reduced failure resistance (*for example.*, a PHTS chemistry excursion which resulted in uniformly high pD), the affected elements would be distributed uniformly throughout the reactor. Therefore, it would be possible to tolerate 1000 elements with reduced failure resistance if the elements were uniformly distributed across the core.

It is possible that sufficient information may exist to deterministically define the location of the elements with reduced sheath failure resistance. If these elements can be shown to be in low power and/or low burnup locations, it may be possible to establish with further analysis that a larger number of elements with degraded sheath failure resistance could be tolerated.

4.0 Analysis Envelope

There is also a need to define the range of fuel modelling input parameters, both dimensionally and for materials, over which we will ensure that there is analysis of the behaviour of fuel for all safety analyses. A review of technical specifications and manufacturing production procedures was performed and values for fuel manufacturing parameters chosen or recommendations made to ensure that the tolerances in manufacturing would be covered in the analysis.

4.1 Methodology

A review of data from a variety of sources, including design documentation, manufacturing documentation, previous safety analysis input files, and measurements, was performed. Judgments had to be made about what values should be used because of inconsistencies in the data and the desire to provide additional operating margin for fuel performance and safety assessments. Where judgments are made they are based on previous experience and may require revision in response to new information.

Note that the specification of fuel is complex and, at this time, we are not able to determine fully how safety analysis should address all the variations in as-built fuel. For example, it was not possible to define an analysis envelope for the yield strength of Zircaloy. However, it was felt that there was sufficient gain in understanding and evaluation of fuel modelling parameters to start this process.

4.2 Dimensional Input

Dimensional data was reviewed and summarized as illustrated in the sample section below. Here, the Column "Nominal" gives values given in the Fuel Design Manual as the standard or typical values. The column "Design Range" was the range of values as specified in the Technical Specifications. The 1992 NBP Tender Document was used here to extract these values. The column "Manufacturing Range" was extracted from manufacturer drawings and is restricted to dimensions only. The column "Analysis Range" was selected by the authors to account for the range of all of these values plus any values previously used in safety analyses and represents the recommended range to be used in future safety analysis.

Parameter	Nominal	Design Range	Manufacturing Range	Analysis Range
Geometry				
Std. Pellet Diameter (mm)	12.15		12.2 +0.02 / -0.008	see Note 1

The manufacturing tolerances are tighter than the accepted design range and since the analysis range should be at least as large as the design range these were not restrictive. In this case, the analysis range was not applied uniquely to this parameter but to the collection of pellet and sheath dimensions. "Note 1" was used to explain this.

Note 1:

The tolerances or ranges of diameters or radial values for pellet, gap, sheath thickness, and pellet outside diameter cannot be specified individually. They must be defined as a consistent set of values. The following 4 sets below are chosen since they represent the extremes or full range of fuel element diametrical dimensions.

	Fat Element + Small Gap	Fat Element + Large Gap	Thin Element + Small Gap	Thin Element + Large Gap
Sheath OD (mm)	13.15[a]	13.15	13.05[b]	13.05
Sheath tk (mm)	0.437	0.37[c]	0.3875	0.37
Diametrical Clearance (mm)	0.025	0.260	0.025	0.160
Pellet OD (mm)	12.25[d]	12.15[e]	12.25	12.15

[a] Rounded up slightly too greater than design to provide a small analysis margin.
[b] similar margin is applied as in footnote 1.
[c] This is 0.01 less than the spec. to provide analysis margin.
[d] This has been rounded up slightly for margin.
[e] This value is the one reported as the nominal OD.

Another example in which the analysis range is slightly larger than the design or manufacturing range is shown below for the endcap outside diameter. The reason for this was explained in an accompanying note.

Endcap O.D.		13.246 +0.0 / -n.r.	13.23 +0.0 / -n.r.	13.12 ± 0.13 (see Note 5)

Note 5:

For the endcap OD and fuel support height the range of analysis values was expanded to envelope the values used in the Fuel String Compression analysis as reported in PLGS-IR-03553-04.

4.2 Material Specifications

This section considers variations in such properties as pellet densities, pellet conductivities, sheath strength, pellet cracks and pits, etc. This is a more difficult subject to address. The note below gives an example of how this was proposed to be dealt with.

Note 7:

The assumption on these material properties is that the variation in materials used at the time of fabrication do not alter the thermal properties (conductivity and heat capacity) by more than 10%; and the mechanical properties (yield strength, strain rate) by more than 10% over the range of all safety analysis.

The equivalent boron concentration (EBC) of impurities should not exceed the values in the technical specifications - these are 1.3 mk for natural UO_2 and 1.6 mk for depleted pellets. Note that a change in the EBC will affect the flux distribution in a pellet and hence also the fuel temperatures.

4.3 Chemistry Envelope

Figure 2 shows that the filling gas volume in an element is fairly consistent between 2 ml to about 2.8 ml. The plot shows for each production run the mean of the samples as well as the minimum and maximums measured. The minimum observed was 1.95, the maximum was 2.88 and the average is 2.24 ml. These volumes are larger than previously used in safety analysis, see Table III-1 of Appendix III of the Safety Report. But this range is smaller than is proposed to be allowed for in the analysis range.

The Helium fraction in the filling gas volume is shown in Figure 3. The minimum observed was 75.3%, the maximum was 97.3% and the average was 89.0%. Traditionally a fill fraction of 90% has been used. This is consistent with the data reported here and will continue to be used.

Figure 4 shows the measurements of stoichiometry of oxygen to uranium ratio. Since the amount of oxygen in the fuel affects it conductivity this value is quite important. The AECL TS-xx-37351-1 is from 1.995 to 2.015. It is seen that our fuel meets the AECL technical specification and therefore the value 2.0 is appropriate for fuel modelling purposes, with sensitivity analysis as appropriate to assess the impact of variations within the design range.

4.4 Application

Five ELESIM input files were created that represented the nominal or centre-line case plus 4 extremes. Since variations in material properties and chemical properties can be imposed upon these base cases there are a large number of combinations. Guidelines were provided on what the range of conditions the modeller should select depending on the objective. That is to predict minimum or maximum gas pressure, minimum or maximum fission gas release, or minimum or maximum fuel element centerline temperature.

The sensitivity anlaysis performed will be used to try and demonstrate that NB Power's application of a bundle mass limit is not required to maintain safety margins, and that these can be addressed with more conventional means of manufacturing controls.

Some guidelines and data were provided for modelling accident transients with ELOCA, although this is mostly driven by the nature of the scenario under consideration.

5. Fission Product Release Code

To upgrade the toolset for use in fission product release modelling, NB Power purchased the right to use of SOURCE 1.0 and joined Ontario Hydro and Hydro Québec in a User Group for the development of SOURCE 2.0. Since that work has been reported upon elsewhere, see reference [9], this will not be discussed here.

6. Outstanding Issues

i. Will need to convert the input files created to those accepted by ELESTRES-IST and repeat the sensitivity analysis mentioned here.

ii. It is not clear if the range in the fuel specifications provide sufficient axial clearance in the fuel that we purchase. It appears that elements are manufactured typically with a 1 to 4 mm internal axial clearance. But the thermal expansion of the fuel would appear to quickly fill in this clearance and leave little space for pellet expansion.

iii. Sensitivity analysis with ELESIM showed that fuel surface roughness was an important parameter. It was felt that this warranted review of the manufacturing control of this parameter with the expectation that the analysis range proposed might be reduced.

iv. The maximum element hydrogen content is specified to be 1 mg. We appear to exceed this, although only slightly, on a regular basis, see Figure 5. Since this appears not to have created any operational problems perhaps this limit should be re-reviewed.

v. The proposed analysis range allows for a much larger variation in internal filling volume than is consistent with the manufacturing range. This suggests that perhaps the analysis range is too large and will need to be narrowed to be more consistent with our as-built fuel.

vi. It appears that the traditional ELESIM input case as in Table III-1 of Appendix III of the Safety Report is a reasonable representation of an average fuel element. But it does not represent the limiting parameters for predictions of internal gas pressure or highest fuel temperatures. However, based on some preliminary sensitivity analysis performed with ELESIM, it is expected that no existing safety analysis would be compromised by the present range of manufacturing specifications and tolerances.

7. References

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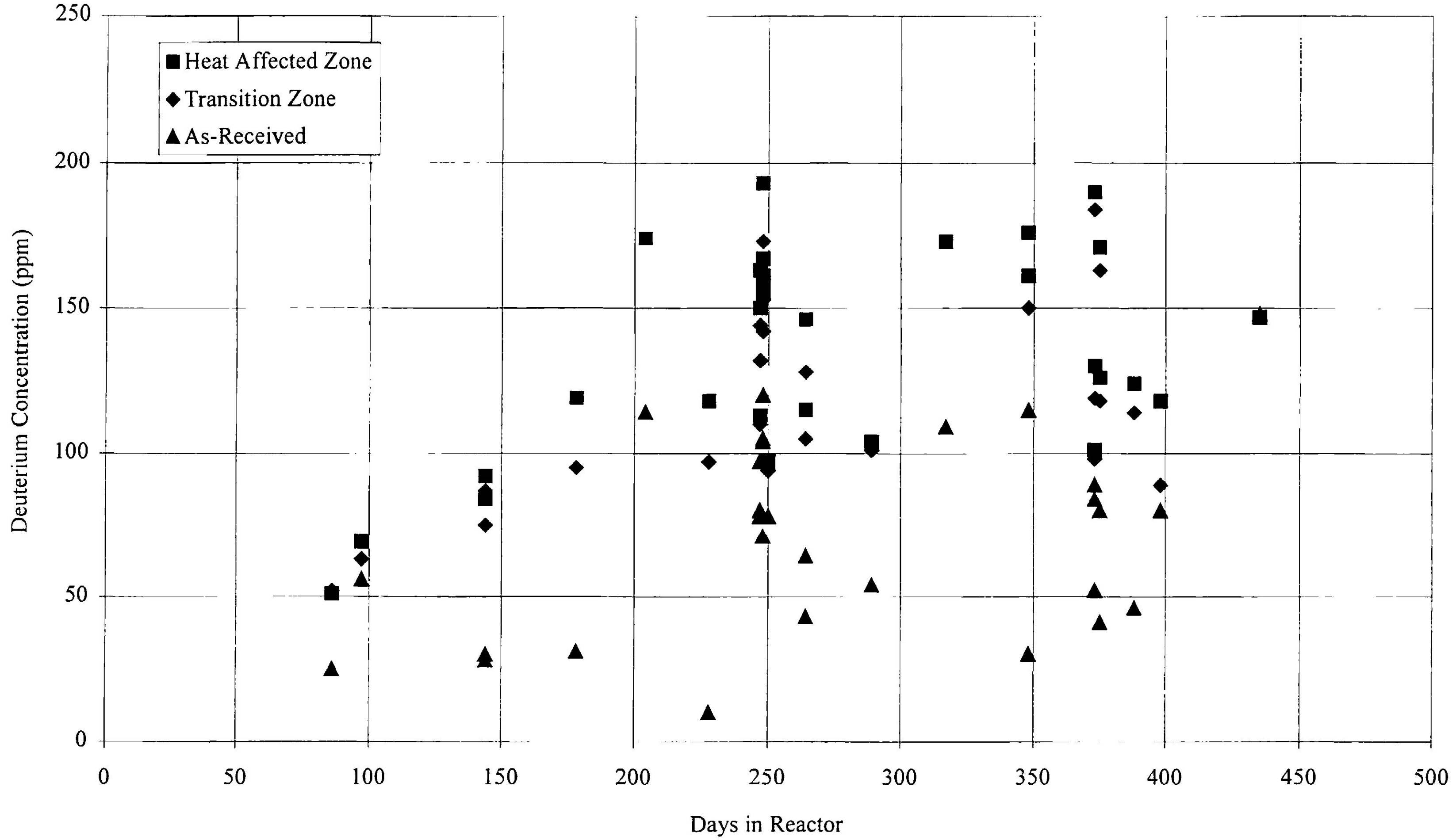

Figure 1: Deuterium Concentrations Measured in PLGS Fuel Elements

Figure 2: Measured Element Gas Volumes

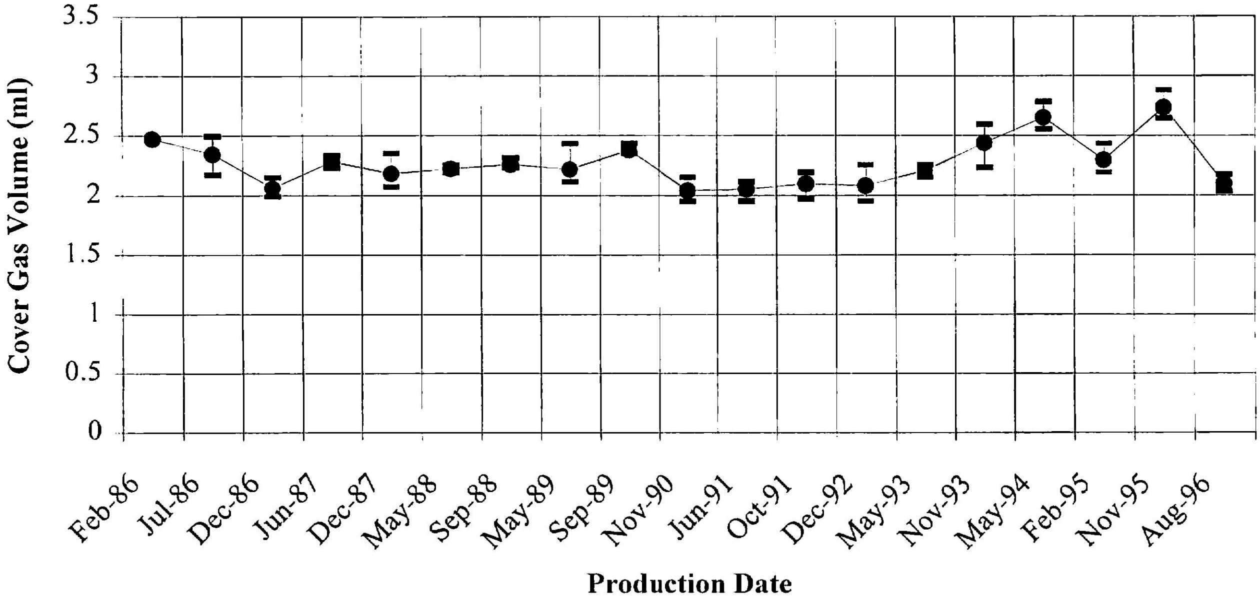

Figure 3: Measured Helium Fraction in Element Gas

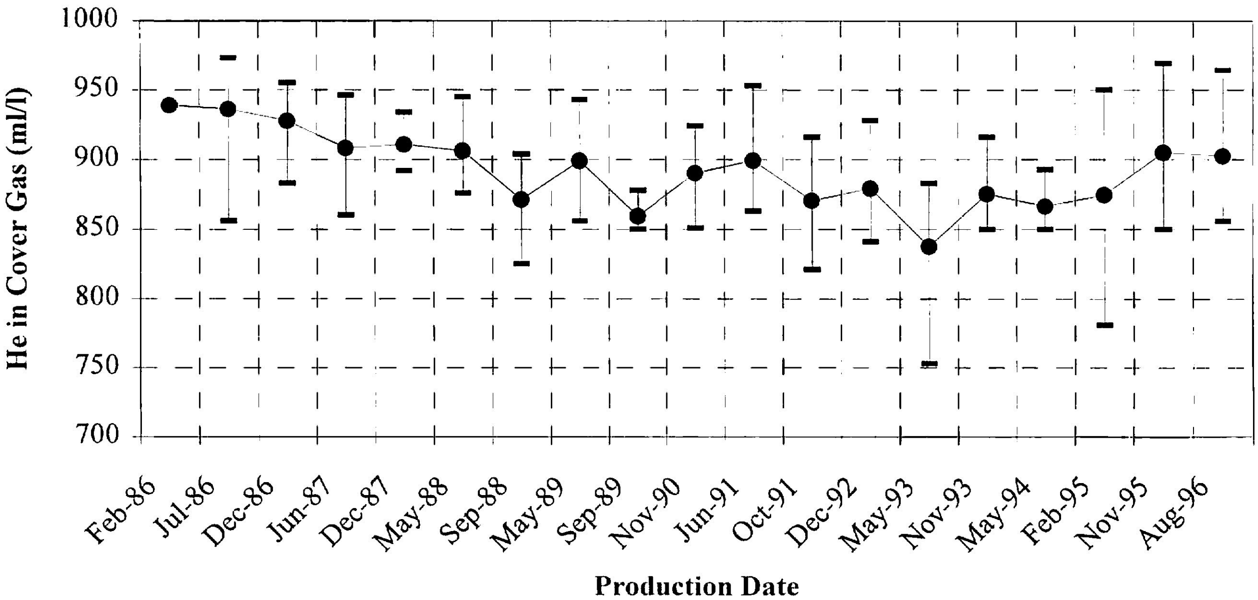

Figure 4: Measured Pellet Stoichiometry

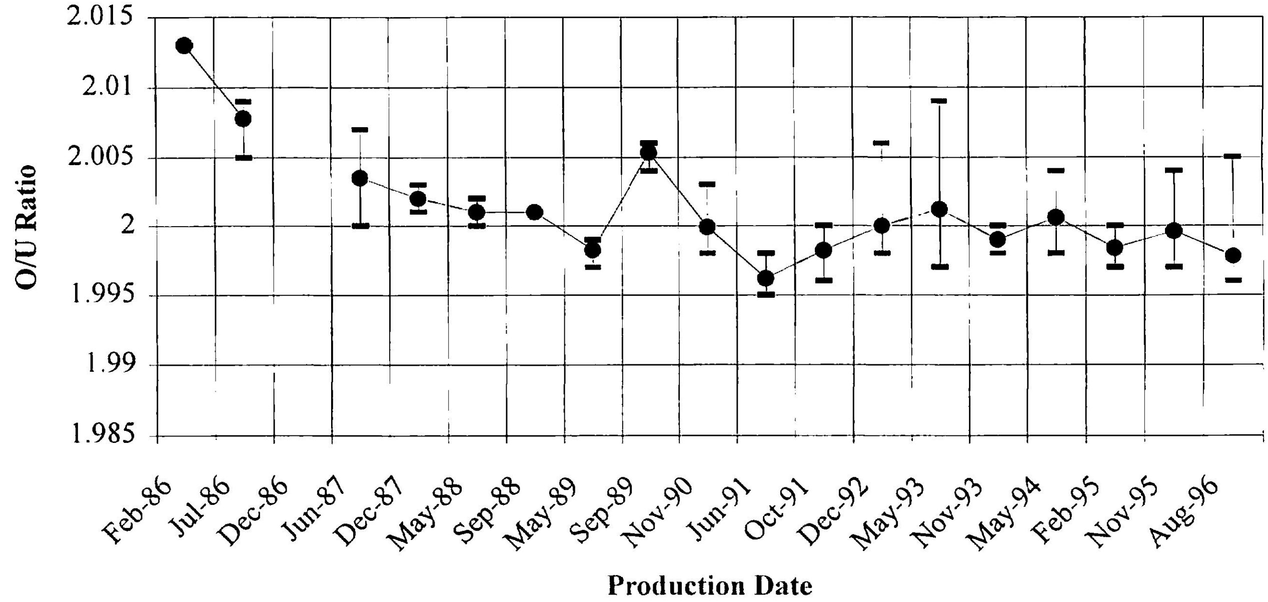

Figure 5: Measured Total Element Hydrogen Content