Operational Support of a Safe Operating Envelope for Fuel

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1.0 Abstract

The mandate of a station safety analysis group is to ensure that the station is operated and maintained in a manner consistent with the basis for our understanding of the safety consequences of process or human failures. As operating experience has developed an awareness of the significance of fuel manufacture and operating conditions on safety consequences has also grown. This awareness has led to a program that is designed to ensure that these influences are appropriately considered. This paper describes the projects that make up this program.

2.0 Introduction

2.1 Background

In 1991 Point Lepreau Generating Station (PLGS) experienced a higher than usual rate of fuel defects. After a concerted effort by the station staff, the fuel manufacturer and Atomic Energy of Canada Limited (AECL) staff, the manufacturer identified a problem with under-baked elements and the likelihood of hydrogen internal contamination as the possible cause of the excursion. In *1991*, several D_2O supply hoses became swollen and white deposits were observed on the outside of the hoses. The cause was identified as hydrogen build-up in-between the layers of the hosing and was due to an inadvertent excessive addition of hydrogen to the Heat Transport System during a routine hydrogen addition procedure.

This event created an awareness of the potential for a change in operating conditions to impact on the basis for safety analysis case for the station.

2.2 Summary

At Point Lepreau a program is under development to refine our understanding of the operating envelope for fuel and to make the safety analysis consistent with this envelope. The definition of an operating envelope for fuel starts with the fuel design as documented in the fuel Design Manual (DM) and the fuel technical specifications. The operating envelope is generated by the range of fuel manufacturing tolerances about the design centre, both with respect to element and bundle dimensions and with respect to the material tolerances. In addition, the rare but occasional fuel manufacturing concessions represent small numbers of bundles with larger deviations in design. The activities taken by NB Power are outlined in this paper

That envelope is then expanded by the affects of in-core irradiation, exposure to HTS pressure and flow, and the range of HTS operating chemistry. Power conditions, both steady state operating powers and power ramps also affect the condition of the fuel. The power/burnup envelope of the fuel bundles is a critical parameter for judging the condition of fuel. To understand the range of conditions of fuel bundles during in-core irradiation, extensive studies have also been performed by Ontario Hydro and AECL.

To support the development of an analyzed envelope of operation for fuel, fuel inspections are performed. A review of our historical fuel inspection process was performed. As well, NBP have inspected all fuel bundles defuelled from the core pass downstream of HTS header 2 to check for debris remaining from the foreign material incident in October 1995. Although the primary intent of these inspections was to provide assurance that no pressure tube damage occurred as a result of the foreign material, they also provide fuel inspection data on a large number of fuel bundles. From this, a fuel inspection program has been refined and routine fuel inspections have been committed to

Once the fuel operating envelope is defined, the safety analysis should be shown to address these conditions. As well, where practical, the safety analysis is extended to show the sensitivity of safety consequences to small perturbations beyond the defined operating envelope.

3.0 Definition of the Fuel

3.1 Fuel Design Manual

The description of the 37-element fuel presently used at PLGS is governed by the current Fuel DM [1], which contains a listing of the drawings and technical specifications which define the fuel. The Fuel DM is then used to generate the Tender Document that is prepared by NB Power staff to convey to potential manufacturers the specifications of the fuel bundles required by PLGS. The potential manufacturers then prepare product specifications which detail the procedures that will be followed in the manufacture of the fuel along with the specifications of the final product delivered to the station. After a review of this document by the station staff, the manufacturer issues a manufacturing and qualification plan which provide details of the manufacturing campaign. After this document is reviewed and accepted the fuel is then manufactured.

The most recent revision of the Fuel DM was in 1982, and this document references Technical Specifications all dated before 1978. In the most recent fuel tender document, 8 of the 11 Technical Specifications were dated after 1978, and there is presently work going on under Working Party 9 to further update some of the specifications. Table 1 identifies the differences between the specifications identified in the Fuel DM and the current NB Power Tender document (which the fuel is presently being manufactured to). Many of the differences between the specifications have been made to reflect manufacturing techniques and the availability of materials. A detailed comparison of all the Technical Specifications was not performed at this time.

In addition, the Fuel DM refers to the original 37-element bundle design drawing, while the tender document refers to revision 4 of this drawing. The fuel is then manufactured to a different drawing, the manufacturer's drawing. The original 37-element bundle design drawing (XX-37000-1-1-GA-E, Rev. 1) referred to in the DM [1], was compared to the drawing (XX-

37000-1-1-GA-E, Rev. 4) currently referred to in the Tender Document [2] with the discrepancies presented in Table2..

Due to the updates to the technical specifications (both in the past and the future alterations) and the design drawing the above review shows the need for the Fuel DM to be updated to ensure that the fuel loaded at PLGS meets the requirements as set out in the DM. The alterations are necessary to ensure configuration management control.

3.2 Bundle Mass Limit

A limitation was placed on bundle Uranium mass due to a Critical Channel Power (CCP) evaluation which showed that a channel with an average bundle mass greater than 19.25 kg U would have a net positive sheath strain, and according to some of the analysis which has been done, a reduction in CCP would result. This analysis was later found to be based on erroneous sheath strain data, which was redone [3].

In an effort to eliminate the bundle mass limit, the literature describing the effects of U-mass on the bundle's performance [4],[5],[6] were reviewed. The results of that search were then compared with the assumptions used in the safety analysis for the station. The two COG funded papers ([4],[5]) showed that there would be no affects to CCP as long as the bundle uranium mass remains below 19.4 kg U. It was also shown that the average uranium mass for a 37-element bundle that is manufactured within the design specifications would be 19.3 kg U with a ±3 sigma range from 19.27 to 19.33.

A further concern that arose from the bundle mass limit was that the vvariations in the initial sheath strain would affect the probability of fuel sheath failure in a Large Break LOCA. This concern was also addressed by the following argument.

In a Large Break LOCA, fuel sheath failure does not occur until the fuel sheath has exceeded at least 800 °C. This assertion can be supported, not only based on the experience which has been garnered from detailed thermal-mechanical modeling using the ELESIM/ELOCA code suite, but also based on the fact that early, bounding safety analyses of Large Break LOCAs used the time that the first fuel sheath was predicted to reach 800 °C as a predictor of failure timing.

The temperature at which the α/β phase transformation occurs is 800 °C. This phase transformation, accompanied as it is by a re-ordering of the lattice structure of the Zircaloy, will eliminate the internal stresses due to the residual manufacturing strains and the initial strain, removing the material's "memory" of it's stress history. Therefore, the initial strain will have no effect on the fuel sheath failure probability during a Large Break LOCA.

At the present time work is progressing to justify eliminating the bundle mass limit by applying standard fuel design limits.

3.3 Monitoring of Manufacturing Parameters

To ensure that the manufacturing steps produce a product within the specified limits in the tender document the following programs are in place:

- The manufacturer issues an Inspection and Test Plan which is then reviewed by PLGS staff
- An NBP Quality Assurance Representative (QAR) is sent to the manufacturing location during bundle fabrication.
- NBP's QAR inspects the first 10 bundles from the manufacturing campaign. These bundles are loaded into the reactor first to provide the station personnel with advanced performance information (the Lead Bundle Program).
- Archive bundles are set aside from each manufacturing run.

- An independent gas analysis is performed on one element for every 500 bundles produced. 11 Properties of the elements are reported including; cover gas volume (Figure 1), hydrogen content (Figure 2), density of the pellets (Figure 3) and U mass (Figure 4).
- An Annual Fuel Performance Report examines all aspects of PLGS fuel including trends in manufacturing parameters [7],[8],[9].

A more complete description of these processes is presented in [10].

4.0 Operational Affects on the Fuel

4.1 Shipping and Fuel Handling

In order to ensure that the state of the bundles is not altered in the shipping from the manufacturer to the generating station, care is taken in the transportation of the bundles. The most significant concern during this period is that the geometry of the bundle may be affected by a sudden jar, compromising its performance in-reactor or its ability to successfully interface with other systems, such as the fuelling machine. Care is also taken when handling the bundle not to alter its condition in any way. The bundles are also visually examined before being placed in the core. A more complete description of this process is presented in [10].

4.2 Heat Transport System

Another function of the Fuel DM discussed in Section 3.1 is to define constraints upon other systems in the plant to ensure that the fuel is operated within its design envelope. The results of a review of these constraints and how they are treated by station documentation at PLGS is presented in Table 3. There were 3 main concerns that were found during this review and the preparation of the annual fuel performance reports [7],[8],[9]:

The discrepancy between the Fuel DM [1] and the Chemistry Control Operating Manual [11] was noted in the 1996 Fuel Performance Report [8] (refer to Figure 5). Presently AECL is reviewing the requirements for pH. When the review is complete, the range in the Fuel DM will be corrected as needed [14].

The fuel DM [1] states that the maximum flow rate during normal operation should be 26.7 kg/s. Flow verification results (normalized inverse heat balance flows) performed on April 4, 1997 [12] and November 22 [13], indicated a maximum flow rate of 30.6 kg/s and 30.4 respectively (under normal conditions and the reactor being at 100% nominal full power). The flow verification also indicated that 130 (04/04/97) and 133 (11/22/97) channels had flows in excess of 26.7 kg/s. Although this flow may not be impacting on all of the bundles (pressure tube creep is leading to flow bypass in the central bundles) the maximum flow rate reported was above the design maximum flow rate. A letter was forwarded to the designer outlining this concern after the 1996 Fuel Performance Report. The reply stated that the Fuel DM should be revised to reflect the new flow regimes [14].

The final discrepancy between constraints on other operating systems made in the Fuel DM [1] and the PLGS documentation on those systems was the treatment of bundles stuck in cross-flow. The Fuel DM specifies a maximum time to be 10 minutes, where the station documentation indicates that a bundle stuck in crossflow for less than 1 hour is unlikely to damage the bundle [15]. This issue will also be addressed inn the production of the new Fuel DM.

4.3 Condition of the Pressure Tubes

The Pressure Tubes (PT) at PLGS are periodically examined by the Central Nuclear Services Group of Ontario Hydro. The analysis is performed using a channel inspection and gauging apparatus for reactors (CIGAR). The results of these examinations are presented in a series of PLGS information reports [16].

5 Monitoring Bundle Condition

In order to ensure that the fuel bundles are behaving as expected in the core, it is necessary to examine irradiated fuel. Presently, PLGS can examine the surface of the spent fuel in the reception bay and the bundles can be sent off for a more extensive Post Irradiation Examination (PIE) at Chalk River Laboratories (CRL).

5.1 In Bay Fuel Inspections

The facility at Point Lepreau provides a periscope with local lighting for visual examinations and has an attachment so photographs can be taken. The bundles are loaded onto the bundle rotator which enables the operator to rotate the bundles about the element axis, move the bundle back and forth so that the entire length can be viewed and to rotate the bundles end-to-end so that the end plates and inter-element spaces can be viewed. These examinations are commonly referred to as Visual In-Bay Examinations or VIB's.

Any abnormalities are recorded along with the bundle serial number, date examined, date discharged, channel and position in the fuel bundle examination report [17]. During the examination photographs are taken of any abnormalities with a description and record of the photos taken included in the report.

The bundles that are VIB Inspected at PLGS fall into one of three categories; the bundle is a defect, the bundle is a sample of 'healthy' fuel, the bundle resided in a channel fed by header 2-3.

All bundles, which are suspected of being defective, are examined in the reception bay. Any noticeable defects are noted and the bundle is set aside for canning. Since PLGS has been in service there have been 57 bundles that have fallen into this category.

Up to 1995, PLGS had performed 190 VIB inspections on 'healthy' fuel, representing 0.25% of all bundles irradiated to that point. All the inspections performed have used standard reporting sheets to record the observations. These are then independently reviewed for signs of abnormal behaviour. As part of our response to Action Item 94-G02, it is proposed that PLGS will VIB inspect 21 'healthy' bundles from various channels and positions per year.

The VIB inspections of all of the bundles that were present in channels fed by HDR 2-3 when the wooden hatch cover event occurred was completed in 1997. 1138 bundles were discharged from channels fed D_2O coolant by HDR 2 and all of those bundles have been examined. One defect has been directly attributed to the debris in the core ([18] Figure 6) and there has been some evidence of debris caught in the bundles being removed from the core without incident (Figure 7). There has been a commitment made to continue to examine all pass 2-3 bundles.

A separate review of the header 2-3 VIB inspections was performed to assess the value of the inspections as fuel inspections and to determine if the condition of the bundles would have an affect on safety analysis assumptions. Bundles that had a noticeable mark or something that might make the bundle more susceptible to failure during a small LOCA were identified (see Table 4 for a listing of these bundles and Figure 8 for an example). If it is assumed that 1/3 of the element surface area of each bundle is visible during a VIB inspection, 0.044% of all elements inspected have a potential for failure. This implies that 72 bundles in the whole core (1.58%) could have had a single element failure. Thus the fission product source term would be larger than assumed in our analysis as reported in the Point Lepreau safety report.

5.2 PIE / Hot Cell Inspections

Three shipments of irradiated fuel have been made. The first was made in February 1987 during which 29 elements were sent including 3 defected elements. A white coating had been observed in the storage bays that appeared to be correlated with the last four bundles in the channel. The elements were sent primarily to examine for the potential for sheath oxidizing. The second shipment was sent in December 1991. Twenty-six elements, 17 defective and 9 intact, were sent as part of the program to identify the cause of the high incidence of fuel defects. The third shipment was sent in October 1994. Two full bundles were shipped to Whiteshell Laboratories for examination as part of the industry program to develop and demonstrate the use of T-shaped bearing pads. A shipment of two bundles that were specially manufactured to contain some elements with out CANLUB ('Notley Bundles') is scheduled for the fall of 1998.

These tests allow for a more complete analysis of a bundle, aid in the determination of defect causes, ensure that the fuel is behaving as expected in the core and help in the development of CANDU fuel. Additional examination capabilities are also under development at AECL, usually funded through the CANDU Owner's Group, to increase the amount of information that can be obtained from in-cell examination of irradiated fuel.

6.0 References

1 I.E. Oldaker, "Point Lepreau Generating Station Fuel Design Manual", DM-87-37000, Revision 2, April, 1982

2 "New Brunswick Power Corporation Nuclear Fuel Fabrication", Tender Document No. 97-0811JS, Aug., 1997

3 M.R. Floyd, I.A. Lusk, "Verification of Past CRL Profilometry Measurements on Irradiated CANDU Power-Reactor Fuel Elements", COG-93-413, April, 1994

4 S. Palleck, B. Wong, "CANDU Fuel Design Review – Parametric Study to determine Partial Effects of Fuel Manufacturing Parameters on Bundle Mass and Sheath Strain", COG-97-151 (COG WP 0908), April, 1998

5 S. Palleck, "CANDU Fuel Design Review – Bundle Uranium Content", COG-96-540 (COG WP 0908), April, 1998

6 B.J. Wong, "Stress Analysis of PLGS Heavy Fuel: Endcap Region and Sheath/Bearing Pad Region (Revision 1)", Memo to R. A. Gibb, AECL File 91-37400, Setember, 1994

7 P. Reid, "Point Lepreau Generating Station Information Report – Point Lepreau 1995 Fuel Performance Report", IR-37000-06, Rev. 0, July, 1997

8 T. Chapman, "Point Lepreau Generating Station Information Report – Point Lepreau 1996 Fuel Performance Report", IR-37000-07, Rev. 0, July, 1997

9 T. Chapman, "Point Lepreau Generating Station Information Report – Point Lepreau 1997 Fuel Performance Report", IR-37000-08, Rev. 0, April, 1998

10 T. Chapman, "Point Lepreau Generating Station Information Report – Point Lepreau Practices for Maintaining Bundle Health", IR-03553-09, Rev. 0, September, 1997

11 K. MacGibbon, "Point Lepreau Generating Station Operating Manual - Plant Chemistry Control", OM-78210, Rev 3, April 13, 1994

12 T. Whynot to I. Lee, "Start-up Flow Verification", TU 06374, TU 01814, April 17, 1997

13 T. Whynot to J. Detorakis, "Start-up Flow Verification", TU 06374, TU 01814, March 9, 1998

14 R. Sejnoha to R. Gibb, "NBP Comments on Fuel Design Manual and TS, 1997 Aug 28 and Sept 4, TU-08721.A", TU 02340, November 5, 1997

15 D. Murphy, "Point Lepreau generating Station Operating manual – Fuelling machine Operations", OM-35200, Rev. 5, June, 1997

16 R. Griffin, "Point Lepreau Generating Station Information Report – In-Service Inspection Report: Fuel Channel Pressure Tubes", IR-03550-14, Rev. 0, February, 1989

17 H. Mousek, "Point Lepreau generating Station Maintenance Manual – Procedure for Handling Fuel Bundles Required for Inspection from 33100 Header 2", MM-35300-FP5, Rev. 1, May, 1996

18 R. Baker to P. Ahearn, "Assesment of Damaged Bundles from Fuel Channel P14", File No. 31100 CIGAR, Sept., 29, 1997

19 R. Sears, D. Loughead, "Point Lepreau Generating Station Operating Manual – Primary Heat Transport System", OM 33100, Rev. 5, Dec., 1997

20 N. Singh, "Point Lepreau Generating Station Design Manual – Fuel Channel Assembly", DM-87-31100, Rev. 2, Jul., 1981

21 R. Baker, "Point Lepreau Generating Station Operating Manual – Spent Fuel Storage and Handling", OM 35300, R2/5, Oct., 1991

22 T.J. Chapman, "Point Lepreau Generating Station Information Report – Point Lepreau 1997 Fuel Performance Report", IR-37000-08, Apr., 1998

23 R.W. Sancton, "Point Lepreau Generating Station Operating Manual – Fuel Management and Reactivity Control", OM 03103 R 0/2, June, 1998

24 D. Murphy, "Point Lepreau generating Station Operating Manual – Fuelling Machine Operations", OM-35200, Rev. 5, June, 1997

25 D. Wilson, J. McIntosh, "Point Lepreau Generating Station Reference Document – Operating Policies and Principles", RD L03 R 6/1, Dec., 1996

26 K. MacGibbon, "Point Lepreau Generating Station Operating Manual – Plant Chemistry Control", OM-78210, Rev. 5, July, 1995

27 T.J. Chapman, "Point Lepreau Generating Station Information Report – Point Lepreau 1996 Fuel Performance Report", IR-37000-07, Aug., 1997

28 R. Baker, "Point Lepreau Generating Station Operating Manual – New Fuel Loading", OM 35100, R3/2, Nov., 1991

Description	Specification in the Fuel Design Manual [1]	Specification in the Most Recent Tender Document [2]
Core Fuel for a 600 MWe CANDU Reactor	TS-XX-37000-4, Rev. 4, Dec., 1976	TS-XX-37000-4, Rev. 6,
Natural Uranium Dioxide Powder	TS-XX-37032-1, Rev. 0, Apr., 1973	TS-XX-37032-1, Rev. 1
Depleted Uranium Dioxide Powder	TS-XX-37032-2, Rev. 1, Apr., 1973	TS-XX-37032-2, Rev. 1
Uranium Dioxide Pellets for Reactor Fuel Elements	MET-47, Rev. 5, Feb., 1974	TS-XX-37351-1, Rev. 0
Visual Quality Standards for UO_2 Pellets	NP-P-403, Rev. 1, Aug., 1964	-
Zirconium Sheet, Strip and Plate	TS-XX-37353-1, Rev. 2, Mar., 1978	Same
Zircaloy Bar, Rod and Wire	TS-XX-37353-2, Rev. 2, Mar., 1978	Same
Zircaloy Seamless Tubing for Reactor Fuel Sheathing	NP-M-1007, Rev. 4, Sept., 1973	TS-XX-37354-1, Rev. 0
Beryllium Metal for Use in Zirconium Braze Alloy	MET-93, Rev. 1, Dec., 1966	MET-93, Rev. 1, Dec., 1966
Identification of Fuel Bundles for Power Reactors	NP-P-399, Rev. 3, May, 1968	-
Uranium Dioxide Scrap	TS-XX-37032-3, Rev. 1, Apr., 1973	-
Graphite CANLUB Coating on Fuel Sheaths	-	TS-XX-37356-2, Rev. 0, Sept., 1989
Beryllium Brazed Appendage Joints on Reactor Fuel Sheathing	-	TS-XX-37357-1, Rev. 0, Oct., 1981

Table 1: Technical Specifications for PLGS 37-Element Fuel

Specification	XX-37000-1-1-GA-E, Rev. 1	XX-37000-1-1-GA-E, Rev. 4
Angle of Crossed Spacer Pads	min 14°	Min 13°
Bearing Pad Surface Contour	51.69 ± 12.70 R (2.035 – 0.500)	12.0 ± 4.0 (0.47 ± 0.16)
Dimension 'H'	3.18 +0.38 - 0.00	min = 3.05, max = 3.69 *
Note 14	-	Drawing states that the maximum diametral clearance and the axial clearance shall be specified by the contractor
Note 15	Indicates only one design drawing	Indicates design drawings
Additional Specifications noted on Rev. 4	-	TS-XX-37351-1 TS-XX-37357-1 TS-XX-37356-2 TS-XX-37000-4

Table 2: Discrepancies between the Original drawing for PLGS 37-Element Fuel and the drawing presently being used

* There is a detailed description of the measurement of this parameter on Rev. 4 of the drawing that was not present on Rev. 1.

Section of FDM	Fuel Design Specification	Treatment of the Specification in the current PLGS Documentation
6.1.4	Maximum Coolant Pressure During Normal operation <12 MPa	11.75 MPa limit – OP&P Reactor Inlet Header Press = 11.23 MPa [19] Reactor Outlet Header Press = 9.89 MPa [19]
2.1.5 & 6.2.7	PT Wear and Fretting Allowance = 0.064 mm PT Internal Corrosion Allowance = 0.10 mm	Accumulated Wear and Corrosion over a 30 Year Period = 0.20 mm [20] Discussion of a new bearing pad design [20]
2.2.2	Maximum BP Wear = 0.25 mm Minimum BP Height = 0.87 mm	None Found
2.3.2 & 6.3.7	Maximum Loads on the Bundles: In Reactor Service = 5900 N Ram Forces = 7500 N Normal Refueling = 12000 N Refueling assuming a C Ram Failure = 23700 N	Maximum Design Loads [21]: Transfer & Discharge Bay Conveyors = 8900 N Transfer Cart & Storage Tray Con. = 6700 N Ram Calibration data is referenced in MM 35200-FP26 – not in the library system.
4.5.1 & 6.1.1	Maximum Coolant Flow < = 27.4 kg/s (normally)	Max. Channel Flows = 24 kg/s [19] This maximum has been exceeded [22]
4.6.1 & 6.2.2	Maximum Step Height = 0.50 mm	Drawing 87-31100-4-1-GA-E, Rev 3 and associated detailed drawings showed no step heights close to 0.50 mm
4.6.1 & 6.2.6	Maximum Gap Length = 20.0 mm	Drawing 87-31100-4-1-GA-E, Rev 3 and associated detailed drawings showed no step heights close to 0.50 mm
4.6.2 & 6.3.12	Minimum Coolant Temperature for Bundle Sliding = 150 °C	PHTS Temp. (Hot 0 Power) [19] = 260 °C PHTS Temp. (Full Power inlet) [19] = 266 °C PHTS Temp. (Full Power outlet) [19] = 310 °C No statements on limiting fuelling to cooling during shut down
4.6.3 & 6.3.9	Maximum Time in Crossflow = 10 mins	[23] Bundles Stuck in Crossflow from 4 hrs to 4 days could lead to endplate, weld damage and fretting (increase in defect probability) [24] States that damage to bundles in Crossflow for less than 1 hr is unlikely and outlines the following procedures: 10 min – Fuelling Eng. Notes Duration & Pos. 4 hrs – Notify Station Manager 24 hrs – Think of Shutting Down
4.6.6 & 6.3.10	Maximum Air Exposure Times: 30 min if subcooled by flooding 5 min if subcooled by sprays 2 min normal operation	4 min – Spent Fuel Alarm [24]

Table 3 continued on next page…

Section of FDM	Fuel Design Specification	Treatment of the Specification in the current PLGS Documentation
4.6.6	Adjacent Magazine Stations should not Contain Irradiated Bundles	[25] Section: 3.07.3
4.7.1	Bundle Power: 99% of all Bundles Within Ref. Envelope 90% of all bundles Within Nominal Design Envelope	Bundle Power < 935 kW [23] Channel Power < 7300 kW [23]
6.1.2	The Fuel Sheath will Remain Wet During Normal Operation	Adequate Cooling of Irradiated Fuel Shall be Maintained at all Times [24]
6.1.3	Coolant Chemistry: pH – 9.5 to 10.5 Dissolved Deuterium Maximum = 25×10^{-3} $dm^3 D_2/kg\ D_2O$ Dissolved Oxygen Maximum = 50 μg/kg Crud Level Maximum = 100 mg/kg D_20	Plant Chemistry Control [26]: pH – 10.2 to 10.8 - This has been violated [27] Dissolved Deuterium Min = 3 Max 25 ml/kg Dissolved O_2 Max. = 10.0 μg/kg Crud Level – This is not defined in the Chemistry OM but staff assure that it is low [22]
6.2.1	Fuel Passage Restrictions	XX-37000-1-1-GA-E, Rev. 4
6.3.5	Bundles shall be moved in a horizontal direction at all times	[28] For Short Periods of Time the Bundle can be Vertical
6.3.8	The side-stop to Guide Tube Clearances shall be 96.90 min and 100.96 mm maximum.	Drawing 87-31100-4-1-GA-E, Rev 3 and associated detailed drawings
6.3.11	No Torque shall be applied to the Bundle During Fuelling	This appears to be met by the nature of the fuel handling system

Table 3: Treatment of the Fuel Design Manual Specifications in PLGS Documentation

Channel	Position @ Discharge	Bundle	Remarks
A14	12	A86162Z	Element number 10 had a scratch with peculiar staining
C16	11	A91745Z	A large amount of metal debris near element 23, this could cause localised flow blockage
E16	12	A89174Z	Element number 7 was partially collapsed, possibly due to a chip in one or two pellets.
H13	11	A89819Z	An element was partially collapsed, possibly due to a chip in one or two pellets.
L20	12	A94745Z	An element was partially collapsed, possibly due to a chip in one or two pellets.
R16	1	A92498Z	A large amount of deposition and corrosion at the endcap

Table 4: Fuel Bundles that were examined due to the HDR 2-3 event that contained an element with a noticeable blemish

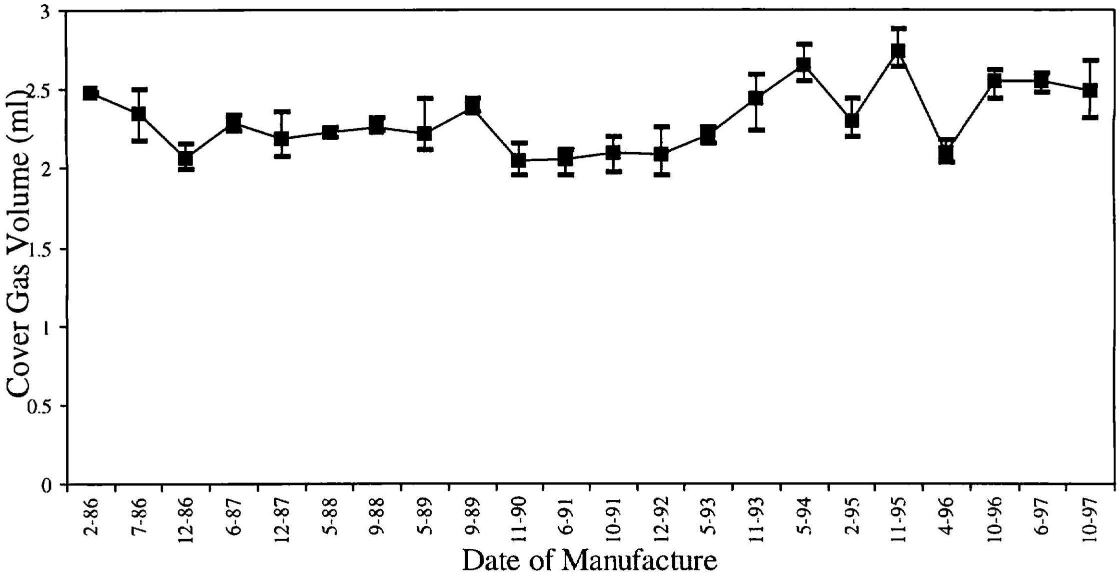

Figure 1: Cover Gas Volume (OHRD Element Analysis Results)

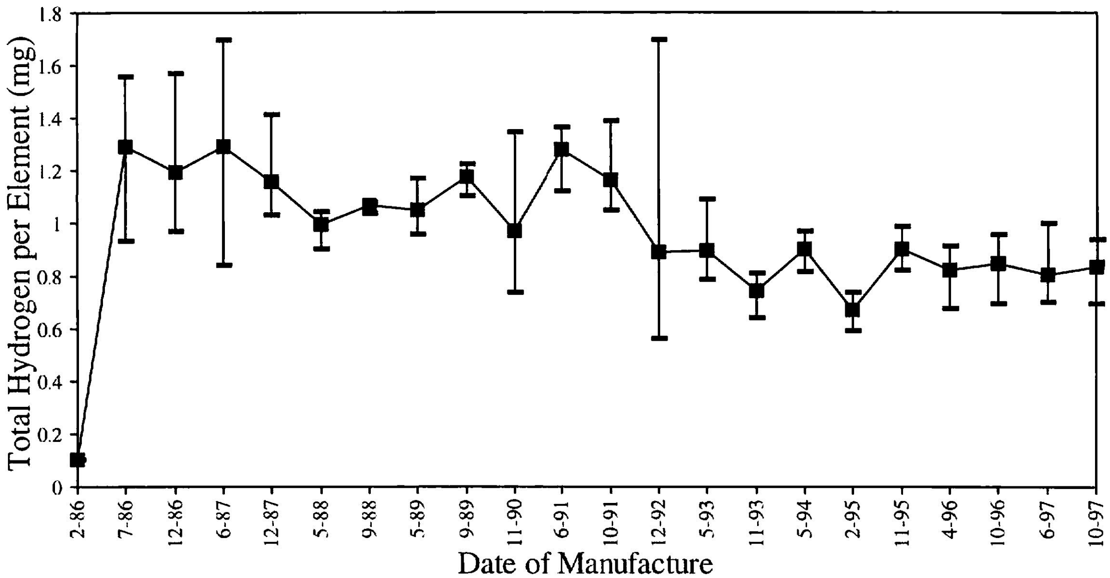

Figure 2: Total Hydrogen per Element (OHRD Element Analysis Results)

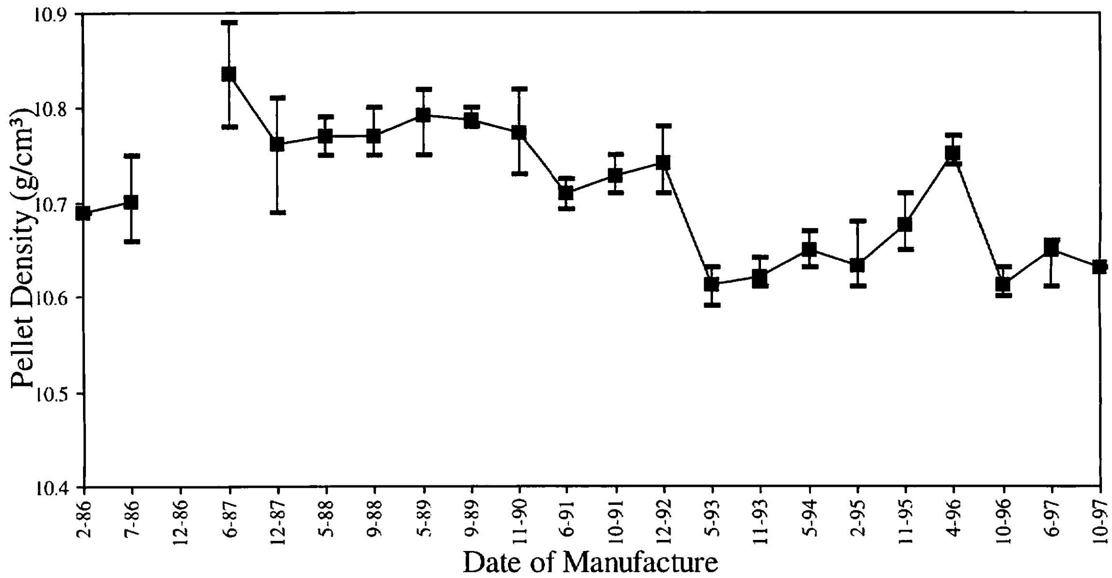

Figure 3: UO_2 Density (OHRD Element Analysis Results)

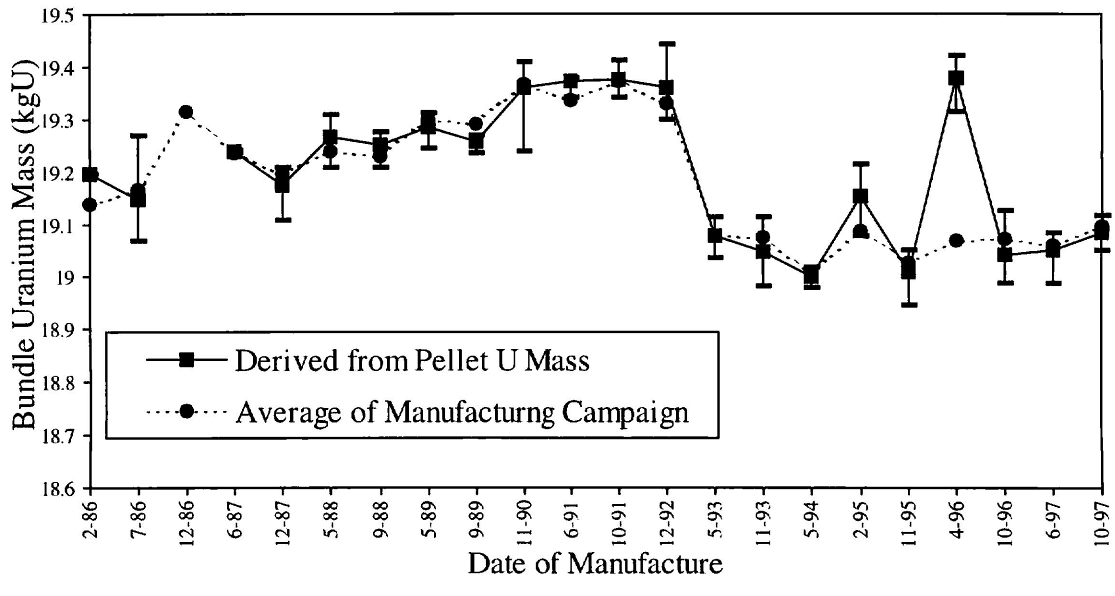

Figure 4: Bundle Uranium Mass (OHRD Element Analysis Results) Comparison of Value Derived from Element U Mass to Campaign Average

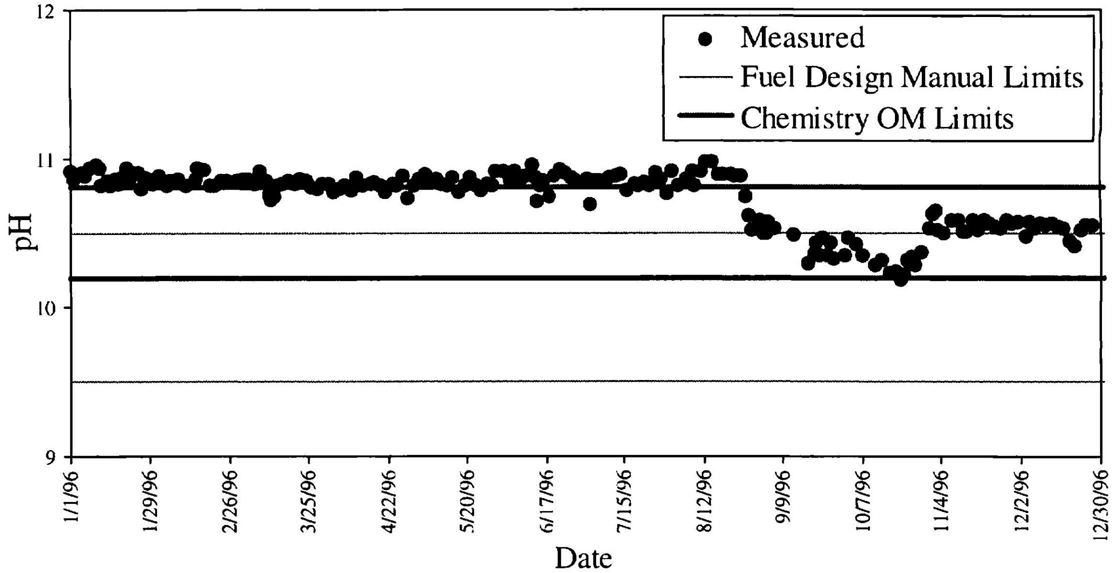

Figure 5: pH of the Primary Heat transport P4 Discharge & Inlet to the PHT Purification Filter FR1/2

Figure 6: Damage to Bundle A96839Z (P14 Position 8) from Debris

Figure 7: Debris between the Endcap and Endplate of Bundle A91532Z (P12)

Figure 8: Bundle A89819Z from channel H13 showing a partially collapsed element, possibly due to a chip in one or two pellets