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ACD3-94SF20218

Technical Overview

CANDU MOX Fuel Dual Irradiation Experiment

Parallex Project

100-37000-TD-001
Revision 0

1996 February

Février 1996

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Release and Revision History / Liste des documents et des révisions

Document Details / Détails sur le document

Title/Titre
Technical Overview - CANDU MOX Fuel Irradiation Experiment

Approval Status / État de l'approbation: A

Total no. of pages / N^{bre} total de pages: 27

Release and Revision History / Liste des documents et des révisions

Release / Document No./N°	Release / Document Date	Revision / Révision No./N°	Revision / Révision Date	Purpose of Release; Details of Rev./Amendement / Objet du document; détails des rév. ou des modif. (A)	Prepared by / Rédigé par	Reviewed by / Examiné par	Approved by / Approuvé par
1	Feb.1996	0	Feb.1996	Approved for use	F.C. Dimayuga M.R. Floyd M.H. Schankula J.D. Sullivan	D.S. Cox	J.I. Saroudis
2	Sept.1996	0	Feb.1996	Issued as contract deliverable D-1			

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. / Proj. conn.	Project / Projet	SI	Section	Serial / Série	Sheet / Feuille No. / N°	Of / De	Codes: Ret. / Rét.	Sec. / Sect.	Dist. / Distr.	Source	Unit No.(s) / Tranche n°
	100	37000	TD	001	1	1					

CDU-939B-00 Rev. 11 95-11

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LIST OF ACRONYMS

AECL	=	Atomic Energy of Canada Limited
CANDU	=	Canada Deuterium Uranium
CRL	=	Chalk River Laboratories
CSA	=	Canadian Standards Association
EDX	=	Energy Dispersive Analysis of X-rays
HFEMS	=	Hot Vacuum Extraction Mass Spectrometry
HPLC	=	High Performance Liquid Chromatography
LANL	=	Los Alamos National Laboratory
MINATOM	=	Ministry of Atomic Energy (Russia)
MITP	=	Manufacture, Inspection and Test Plan
MOX	=	Mixed Oxide Fuel
NRU	=	National Research Universal (research reactor)
O/M	=	Oxygen to Metal Ratio
PIE	=	Post Irradiation Examination
QA	=	Quality Assurance
RF	=	Russian Federation
RFFL	=	Recycle Fuel Fabrication Laboratories
TIG	=	Tungsten Inert Gas
WDX	=	Wavelength Dispersive Analysis of X-rays
WPu	=	Weapons Grade Plutonium

1. INTRODUCTION

Several options are being considered by the United States and Russian Governments for the disposition of excess weapons plutonium (WPu) derived from military programs. The options include possible conversion to Mixed OXide (MOX) fuel and subsequent disposition/utilization in commercial CANDU power reactors operated by Ontario Hydro. The feasibility of the CANDU MOX fuel option was established in a study sponsored by the United States Department of Energy (US DOE) [1]. This feasibility study covered technical and strategic issues, schedule, and cost related parameters, with the objective of identifying an arrangement permitting consumption of 50 tonnes of WPu as MOX fuel in CANDU reactors at the earliest date.

In advance of the final selection of the options favoured by both the U.S. and Russian Governments, it is proposed that a small amount of MOX fuel (rods) be fabricated in each country and irradiated in the NRU (National Research Universal) reactor situated at the AECL Chalk River Laboratories (CRL) in Canada. This would permit evaluation of important technical uncertainties such as the effects of WPu conversion and subsequent processing operations on characteristics of the MOX fuel, and the influence of fabrication methods on irradiation performance. These issues will be discussed briefly in Section 2 (Technical Objectives) and more fully in Section 3 (Scope of Work Description).

This dual irradiation project has been named the Parallex Project; it is a <u>parall</u>el <u>ex</u>periment demonstrating simultaneous disposition/utilization of WPu from the U.S. and Russian Federation (RF). The overall Parallex Project is described in a Project Plan document [2]. This document provides information on project organizational structure, applicable quality program, schedule and cost. A plan for fissile material and equipment transportation in the Parallex project is described in a separate document [3].

The Parallex Project will implement a quality assurance (QA) program in accordance with the requirements of the AECL Management Manual, 00-01914-MAN-014. The QA program described in the Management Manual satisfies the QA requirements of the Canadian Standard Association (CSA) N286 series. The Project will specifically apply the CSA N286.2 standard, Design Quality Assurance for Nuclear Power Plans, and its design verification requirements, to ensure that the irradiation and post-irradiation examination phases of the project, are planned, documented and verified. Procurement documents will address all requirements placed on the suppliers (i.e., Bochvar Institute and Los Alamos National Laboratory for MOX fuel; Chalk River Laboratories Fuel Development Branch Ceramics group and Fabrication, Development & Testing group, as suppliers of enriched and Dy-poisoned fuel elements), including the appropriate QA program standard to be applied to the product or services. The QA program standard specified in the procurement documents is usually from the CSA Z299 series. However, AECL's procedures make it possible to accept other international standards, which may already be implemented by a supplier (e.g., from the ISO 9000 series, the ANSI/ASME NQA-1 standard or Appendix B of 10 CFR Part 50) in lieu of the Canadian standards, provided that QA program equivalence can be demonstrated.

The scope of the Parallex Project is limited to the fabrication, irradiation, and post-irradiation assessment of a small amount (about 70 kg) of CANDU MOX fuel containing up to a few kilograms of WPu. Plutonium derived from excess weapons "pits" will be fabricated into CANDU reactor fuel in both a Russian Laboratory (Bochvar Institute in Moscow) and a U.S. Laboratory (Los Alamos National Laboratory, or LANL, at the TA-55 facility). Four distinct batches of fuel (about 5 to 10 kg each) will be fabricated at each facility, to meet the requirements of specifications supplied by AECL. Each batch would have unique characteristics, tailored to obtaining specific information from the subsequent irradiations. The CANDU MOX fuel will be shipped to the Chalk River Laboratories (CRL) in Canada, where it will be carefully characterized[1], then assembled into demountable fuel bundles and irradiated in the NRU reactor for a period of up to two years. Selected fuel elements will be removed periodically during the irradiation to provide information at different burnup levels. Following irradiation, the fuel will be examined in hot-cells to assess its irradiation performance, and eventually disposed of in a geologic repository. The disposal of all wastes generated during fabrication or PIE activities will be the responsibility of the generator, and disposed of according to applicable procedures and regulations.

Comparisons of the pre-irradiation and post-irradiation measurements, combined with fuel performance data from the irradiations, will be used to determine how production and processing variables, as well as the detailed design of the MOX pellets themselves, can affect the performance of CANDU MOX fuel. The same comparisons would also be used to evaluate specific MOX fuel fabrication parameters, leading to optimization of the CANDU MOX fuel specifications and fabrication methods.

The Parallex Project is a significant first step towards an optimized full-scale WPu disposition program in existing CANDU reactors. However, additional effort would be required to qualify production of WPu MOX fuel bundles for use in the Bruce CANDU reactors. The necessary fuel and physics qualification program is described in Section 2.9 of reference [1]. This effort would include:

- Confirmation of design (demonstration that the fuel design meets all the requirements of normal reactor operating conditions)

    - fabrication and subsequent irradiation of fuel elements and bundles in the NRU research reactor, followed by post-irradiation examination (PIE),
    - reactor physics lattice properties measurements on 35 bundles in the zero energy ZED-2 reactor,
    - critical heat flux measurements on a full-length string of electrically heated bundles in water and in freon.

- Prototype bundle irradiations in Bruce-A

    - fabrication and subsequent irradiation of 20 to 50 prototype MOX bundles in the Bruce-A reactors, followed by selective PIE.

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[1] Characterization in this context includes pre-irradiation inspections and some additional measurements.

The Parallex Project would address the first phase of design confirmation, by providing data on the effects of fabrication parameters on irradiation performance, thereby helping to finalize CANDU MOX fuel specifications and to assess fuel element design features.

This Technical Overview describes:

- the technical objectives and rationale for the choice of MOX fuel fabrication parameters that are to be investigated,
- the pre-irradiation fuel characterization plan,
- the NRU irradiation plan,
- the post-irradiation examination plan, and
- a summary of the evaluations that can be extracted from the Parallex data.

This Technical Overview is based on the 37-element reference CANDU MOX fuel design established in the 1994 Pu Dispositioning Study [1]. An extension to this study is currently underway, aimed at increasing the Pu disposition rates of the mission. The results of this new study will likely specify a higher Pu loading for the CANDU MOX fuel. If confirmed, this Technical Overview document will be revised and the Parallex test matrix could be modified accordingly.

2. TECHNICAL OBJECTIVES

An important policy objective is to encourage U.S., Russian and Canadian cooperation through simultaneous dispositioning and use of weapons-derived Pu in Canada, by commencing an irradiation of CANDU MOX fuel in the NRU reactor in 1996.

There are several important technical objectives for the Parallex Project.

1. The primary technical objective is to evaluate how controlled variations in fabrication processes affect the irradiation performance of CANDU MOX fuel containing WPu. This information will be useful for finalizing the CANDU MOX fuel specifications and defining the fabrication methods to be used for larger scale CANDU MOX fuel production.

   The Parallex Project will provide data that can be used to finalize specifications on impurity concentrations, Pu homogeneity and pellet geometry for CANDU MOX fuel. This will be based on post-irradiation measurements of fission gas release, cladding strains, microstructural evolution and failure thresholds.

   Many aspects of the fuel fabrication process will be important in determining final properties of the MOX fuel. The pit-to-oxide conversion process will affect the processing characteristics of the PuO_2 powder and MOX fuel properties (e.g., homogeneity). The impurity content of the WPu will also be affected by the pit-to-oxide conversion process (e.g., Ga and Am concentrations primarily). Differences between Russian and U.S. MOX fuel fabrication processes (primarily conversion, blending and mixing operations) will result in different properties in the finished fuel. The impurity content and Pu homogeneity will be carefully monitored in the as-fabricated and irradiated MOX fuel, and this information will be combined with measured fuel performance data to infer how fuel performance is influenced by fabrication methods.

2. A second important technical objective of the Parallex Project is evaluation of the CANDU MOX fuel design described in the 1994 DOE study [1][2]. Experimental data will be obtained on the performance limits of CANDU MOX fuel over a wide range of operating conditions. The Parallex irradiations will be done at linear heat ratings both within and beyond the power-burnup operating envelope for MOX fuel in the Bruce-A reactors (as described in the 1994 DOE study [1]). Post-irradiation measurements of fission gas release, cladding strains, microstructural evolution and failure thresholds will be useful in determining performance limits, and will also provide data for validation of computer code calculations of MOX fuel performance.

3. Another key objective of the Parallex Project is to confirm that no unanticipated factors adversely affect the fabrication, handling or irradiation of CANDU MOX fuel containing WPu.

[2] The Parallex fabrication and irradiation program could be modified to include future revisions to the CANDU MOX design.

3. FUEL GEOMETRY AND DESIGN

The reference CANDU MOX fuel design (described in reference [1]) is a 37-element fuelled bundle with depleted uranium throughout (0.2% U-235), containing 5% dysprosium (a burnable neutron absorber) in the central 7 elements (the central element, and the surrounding ring of 6 elements), 2.0% Pu in the third ring of 12 elements, and 1.2% Pu in the outer ring of 18 elements. The bundle average burnup of the reference MOX fuel is 9,700 MWd/te heavy element (HE), compared to 8,300 MWd/te HE for natural uranium fuel in the Bruce-A reactors. Peak element burnup is about the same as for natural UO_2 (about 16,000 MWd/te HE).

Four demountable fuel bundles will be used for the Parallex irradiations in the NRU reactor. The geometry of the demountable bundles is the standard 37-element bundle design, with the following exceptions:

- the central element is removed to accommodate a guide tube required for assembling the bundles vertically in "fuel strings" in NRU,
- the 18 fuel elements in the outer ring of the bundle have special end caps with spigots that permit them to be mechanically locked into the bundle, and
- the bundle end plates are modified to accept these special outer elements and to secure them with a retaining ring.

Figure 1 shows a demountable bundle with the outer elements removed. Figure 2 shows a schematic cross section of the demountable bundles, showing the central guide tube, the core elements in the two inner rings, and the 18 removable outer elements.

The design of the demountable MOX fuel elements will be the same as for natural UO_2 fuel elements currently used in standard 37-element fuel bundles in CANDU power reactors, except that special end caps are used. The geometry of the MOX fuel pellets will be slightly modified relative to the standard Bruce design reference [4].

Two compositions will be used for the fuel in the inner two rings (the core) of the demountable bundles. Three bundles will be fabricated with natural UO_2 (0.7 % U-235) in the core elements, while the other bundle will contain depleted UO_2 (0.2-0.3 % U-235) with 5% Dy added as a neutron absorber. The poisoned core composition is required to provide additional flexibility in controlling the linear powers in the outer elements during the NRU irradiation.

Two Pu contents will be used for the demountable elements: 1.2% and 2% Pu (wt% in HE). This is for consistency with the compositions in the reference MOX bundle design. Preliminary physics assessments indicate that these Pu contents can be irradiated in NRU to produce the required operating conditions (see Section 4.4 - Irradiation Test Matrix).

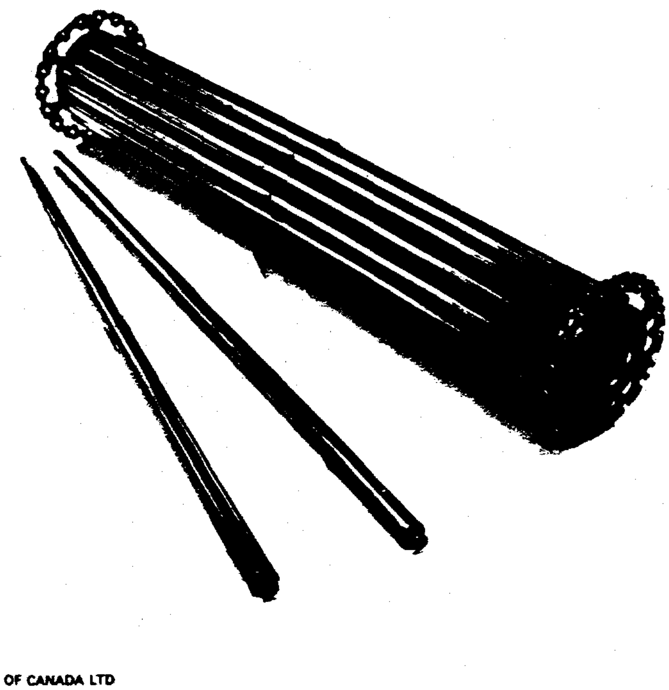

Figure 1: Demountable bundle use for test irradiations in the NRU reactor. The bundle core is shown with the l8 outer elements removed. Two of the replaceable outer elements are shown detached from the bundle.

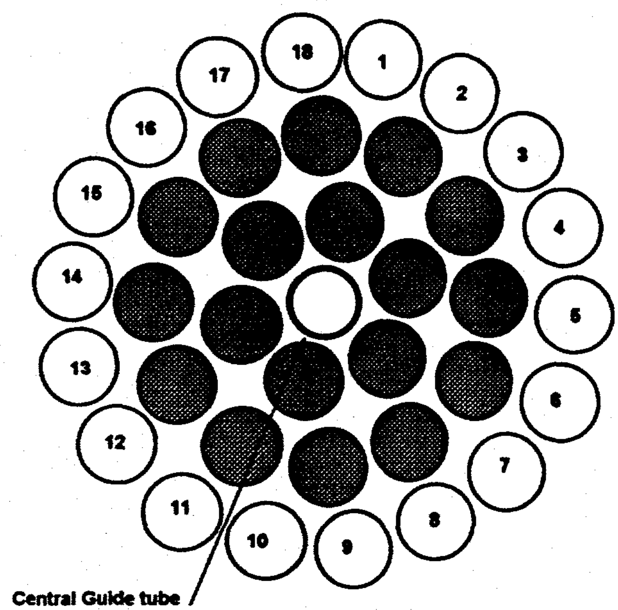

Figure 2: Schematic cross section of a demountable bundle, showing the central guide tube, the core elements in the two inner rings, and the 18 removable outer elements)

4. SCOPE OF WORK DESCRIPTION

The intent of this project is to commence a dual irradiation of CANDU MOX fuel of U.S. and RF origin as early as possible in 1996. The intent is to plan and conduct the fabrication and irradiation efforts so that the information from this project can be used to finalize the specifications and the fabrication methods applicable to larger-scale production of CANDU MOX fuel. Fabrication and irradiation parameters will be controlled so that the NRU irradiations will yield performance data that answer the high priority questions for MOX fuel.

The project technical scope of work can be broken down into the following key activities: Further details of the individual activities are provided in reference [2].

- establish fabrication methods to be used in the TA-55 and the RF facilities,
- produce sufficient PuO_2 derived from weapons sources in each country,
- define an irradiation test matrix,
- establish fuel specifications and QA/inspection requirements,
- fabricate Dy-poisoned UO_2 pellets at CRL for use in a demountable bundle core,
- fabricate enriched UO_2 pellets at CRL for use in control fuel elements,
- provide each facility with the necessary process and inspection equipment,
- qualify the production of acceptable CANDU MOX fuel pellets in each facility,
- produce a first batch of fuel pellets in each facility (about 15 kg MOX, sufficient for about 30 fuel elements),
- ship the first batches of MOX fuel pellets to CRL,
- load and weld the first batch of MOX pellets into fuel elements at CRL, and assemble into a demountable fuel bundle,
- start the dual irradiation in the NRU reactor,
- fabricate three subsequent batches of MOX fuel pellets at both facilities (7 to 10 kg MOX in each batch),
- ship the remainder of MOX fuel pellets to CRL, for assembly into demountable fuel elements and bundles,[3]
- complete pre-irradiation examinations and measurements of the MOX fuel pellets and elements and other control elements,
- irradiate the demountable bundles,
- remove and replace groups of fuel elements on each demountable fuel bundle when target burnups are achieved,
- complete the irradiation,
- complete post-irradiation examination of the MOX fuel elements and other control elements, and
- prepare progress reports for all activities and complete final reports.

[3] If qualification of fuel element welding can be completed at the MOX fabrication facilities, then completed fuel elements would be shipped to CRL, instead of pellets.

It is expected that the entire scope of work would be completed over a period of about 3 years, with a target commencement of the NRU dual irradiation (first batch) in 1996. Details of the schedule and costs are given in the Project Plan [2].

4.1 MOX Fuel Fabrication

The intent is to fabricate CANDU MOX fuel with a limited number of parameters controlled within specific ranges. For the Parallex Project, it is not intended to closely mimic the fabrication methods used currently in operating commercial MOX facilities (i.e., the MIMAS or SBR process), but instead to produce a range of characteristics (e.g., Pu homogeneity) that bound the characteristics of commercially fabricated MOX fuel.

Many characteristics of MOX fuel could be important to its in-reactor performance. Some of the most important MOX fuel characteristics are:

- Surface finish. Surface finish influences fuel performance through its effect on pellet-to-sheath heat transfer. CANDU fuel pellets are normally finished with a wet centreless grinding operation. It is preferable to use dry grinding for MOX fuel because of liquid waste considerations and criticality concerns that often prevent the use of moderating liquids in the MOX glove boxes. It is believed that the nominal specification for surface finish (0.8 μm AA) can be met with dry grinding. If this is not achievable by one of the laboratory fabrication facilities in the time required, the specification might be relaxed and surface finish would be treated as an experimental variable. Otherwise, surface finish will not be a variable in the Parallex test matrix.

- Degree of plutonium homogenization. Although PWR experience suggests that this is not a sensitive parameter for the fabrication methods currently in use, we need to confirm this for the higher linear heat ratings in CANDU[4]. It is expected that sufficient variation in Pu homogenization could be achieved by both laboratory fabrication facilities, such that the specification for production fuels could be evaluated. The homogeneity of pellets from each method will be adequately characterized.

- Gallium content. Ga is of interest because it is present as an alloying element in the WPu. Although no evidence is available to indicate that Ga impurities would be a problem, the Ga may have adverse effects on in-reactor fuel behaviour, such as stress corrosion of the cladding or microstructural evolution of the fuel. Preliminary analysis results from LANL TA-55 pellet fabrication trials and from analysis of PuO_2 derived from WPu indicates that significant amounts of Ga may be removed during pellet sintering operations. Therefore, it appears that Ga concentrations in the finished MOX fuel will be low enough to eliminate preliminary concerns. Ga concentrations will be monitored carefully during pre-irradiation and post-irradiation characterization, but the concentration will not be purposefully varied during MOX fuel fabrication.

[4] The peak ratings of CANDU fuel in Bruce-A are about 55 kW/m, compared to typical LWR fuel with peak ratings of about 30 kW/m.

- Impurity content. Other impurities (contained in WPu - e.g., Am, or introduced during fabrication - e.g., Fe from the steel grinding media) will not be included as experimental variables, since the Pu source is limited to weapons pits that are not expected to be "dirty" and introduction of other impurities during processing is not expected to be significant. As for Ga, other impurities will be monitored during pre- and post-irradiation characterizations, but will not be purposely varied.

- O/M ratio: The oxygen-to-metal ratio will not be a fabrication parameter for this project, but will be monitored during characterization because of its potential impact on fuel performance. Future studies might investigate this parameter.

- Pu content: As noted in Section 3, 1.2% and 2% Pu (wt% in HE) will be used to simulate the reference CANDU MOX fuel compositions. This will not be varied. The depleted uranium feedstock will contain 0.2-0.3 wt% U-235.

Discussions between AECL staff and facility staff at TA-55 and Bochvar are planned. The aim will be to review available equipment in each facility (powder mixing, blending, pressing, sintering, pellet grinding, element welding etc.), and to discuss the proposed fabrication methods to minimize additional equipment procurement, while meeting project needs.

Discussions with TA-55 staff have identified that it would be feasible to produce two levels of Pu homogeneity in the MOX fuel using existing equipment. In both cases, a master mix of 10% PuO_2 in UO_2 would be high-intensity vibratory milled, then blended down to the required composition of 1.2% Pu or 2% Pu, and mixed with a V-blender. This method would be expected to produce a moderate level of Pu homogeneity. A final high-intensity vibratory milling of the entire batch would then be done in one case, to produce a higher level of homogeneity. The milling media would be steel balls.

The four batches of fuel from TA-55 will therefore consist of intermediate and high levels of Pu homogeneity at 1.2% and 2% Pu contents.

As of 1996 February, only preliminary discussions have been conducted with the staff of the Bochvar Institute. Early indications are that a single mechanical mixing method would be used; an electromagnetic high-intensity blender with steel needles as the mixing media. This would be used for both the master mix and the blended final mixture. A high level of Pu homogeneity has been achieved by the Russians, for Pu contents above 5% using this process. They are currently investigating oxidative sintering (a high oxygen potential atmosphere in the furnace during the early sintering stage, finished in a conventional reducing environment). The oxidative sintering is claimed to promote Pu diffusion during sintering, leading to extremely high Pu homogeneity in the sintered MOX. The details of how to produce two levels of Pu homogeneity at Bochvar are yet to be determined, but could include two mechanical mixing processes, similar to TA-55, or alternatively, conventional sintering alone and oxidative sintering as a second method. The intent is to produce four batches, 1.2% and 2% Pu compositions, both by two methods. The current glove box equipment at Bochvar does not include a centreless grinder. Hence, the grinding operation may be done at CRL in the RFFL, if this is determined to be more expedient and cost effective than providing the equipment and qualifying the process at Bochvar.

Technical specifications for MOX pellets [4] and fuel elements [5] will be provided by AECL to both laboratory fabricators. A Manufacturing, Inspection and Test Plan (MITP) shall then be prepared by each laboratory fabricator, describing the detailed methodology of fabricating the fuel pellets/elements. It shall outline:

- the sequence of operations that will be followed during the fabrication campaign, plus reference to the fabrication process procedures,
- the characteristics of the product (both in-process and final) to be inspected,
- inspection and test methods, and reference to inspection procedures,
- acceptance criteria for each inspection,
- and the location of inspection points in the manufacturing flowsheet.

The MITP will be subject to review and acceptance by AECL.>

The required MOX fuel pellets will be produced by both laboratory fabricators in four batches, following qualification of the processes. A pseudo-purchase order from AECL will specify the quantities, composition and the required delivery schedule for the MOX fuel pellets/elements.

Two shipments of MOX fuel pellets/elements are planned. The first shipment from each laboratory fabricator to CRL would consist of the first batch of MOX fuel pellets. The MOX pellets would be loaded and welded into Zircaloy fuel element kits at CRL in the RFFL. This would facilitate a rapid start-up of the NRU irradiation. The second shipment from each laboratory fabricator would consist of the remaining three batches of MOX fuel, either as pellets or as finished fuel elements (TIG welded at the laboratory fabrication sites if possible). Details of the material transportation arrangements, including packages and permits are contained in [3].

Criticality approvals would be obtained for receipt and handling of the MOX fuel in NRU, RFFL, and the fuel fabrication (bundle assembly) laboratory at CRL.

4.2 Production of WPu

The actual production of PuO_2 powder from weapons material is not part of the scope of work for the Parallex Project, although it will be an essential activity to be completed before the start of MOX fuel fabrication. It is assumed that sufficient WPu powder will be supplied to the Project by the US DOE and by MINATOM (RF). The quantity of WPu required from each supplier is about 800 g oxide (500 g in finished fuel, plus an allowance for scrap losses during fabrication and qualification trials).

It is recognized that different processes will be used by each laboratory to convert the Pu-alloy material into oxide, and subsequently process the oxide into ceramic-grade powder. It is assumed that the impurity content of the oxide powder, especially the gallium (Ga) content, will depend on the conversion and processing methods used. The PuO_2 powder properties (particle size distribution, surface area, flowability etc.) will also depend on the processing operations. In turn, these will affect the subsequent mixing/blending operations, as outlined in Section 4.1

The ARIES process, developed at Lawrence Livermore National Laboratory (LLNL) will be used for conversion of U.S. material. The process to be used in the RF is not known.

Detailed analysis of the impurity content of the WPu powder will be essential information supplied with the powder. A procurement specification for the WPu will describe the requirements of this analysis.

4.3 Pre-irradiation Characterization

The MOX fuel fabrication campaign will include inspection and characterization of the fuel according to the MITP (see Section 4.1). This will stipulate the following measurements at specific sampling frequencies:

- dimensional measurements on the pellets,
- density measurements,
- impurity analysis in the finished pellets,
- surface roughness measurements,
- O/M ratio measurements, and
- Pu homogeneity assessments by microprobe analysis, and
- pellet quality or workmanship (appearance, cracks, chips, etc.).

Samples will be stored as an archive from each batch and shipped to CRL for retention and subsequent micro-characterization for Pu homogeneity, microstructure and impurity analysis. This additional pre-irradiation characterization is suggested to confirm the results from U.S. and RF archive samples using a common analysis method, since the US and RF characterization will use different methods to measure O/M and Pu homogeneity. A common set of analyses will be important to compare with post-irradiation assessments at CRL. The welded and finished MOX fuel elements will be individually measured for diameter and bow (curvature) using axial profilometry.

4.4 Irradiation Testing

The irradiation performance of the MOX fuel is expected to be very similar to standard fuel, with minor variations due to subtle differences in the MOX fuel pellet characteristics compared to natural UO_2 CANDU production fuel. The intent is to assess irradiation performance over and beyond the expected envelope of fuel operating conditions. Individual demountable elements will be removed and replaced with fresh elements at target burnups of about 5,000, 10,000 and 15,000 MWd/te HE.

Approximately 90% of the bundles in the reference MOX core will have intermediate and outer ring linear element ratings below 40 kW/m; few, if any, bundles will have element ratings above 60 kW/m. MOX fuel performance for three ranges of element ratings (35-40 kW/m; 45-50 kW/m; and around 65 kW/m) will be tested, to determine the effect of element rating (fuel temperature) on performance. Preliminary calculations of linear ratings have been done for the reference Pu contents (1.2% and 2%) in two demountable bundle designs (natural U and Dy-poisoned cores), at three different NRU loop positions (thermal flux levels). In these calculations, the outer ring was assumed to contain both 1.2% and 2% Pu (alternating)

Figures 3-5 show calculated linear ratings for 1.2% and 2% Pu with natural U (NU) and Dy-poisoned demountable bundle cores. These Figures also show the reference power-burnup envelope for MOX fuel in the Bruce-A reactors [1]. The results show that the 2% Pu MOX can operate with the poisoned cores at low and intermediate flux levels in NRU, while the 1.2% Pu MOX fuel can be operated in any flux level with either the Dy-poisoned cores or the natural U cores. Thus, it will be possible to irradiate 1.2% Pu MOX fuel elements in each of the three ranges of ratings, and the 2% Pu MOX can only be irradiated at intermediate and high linear powers. Derating of local power in NRU, or pre-conditioning at low powers will be required to prevent over-powering of the fresh fuel bundles at low burnup (<50 MWh/kgU). This will be done by irradiating at a lower position in the loop (pre-conditioning) at the start of the irradiation.

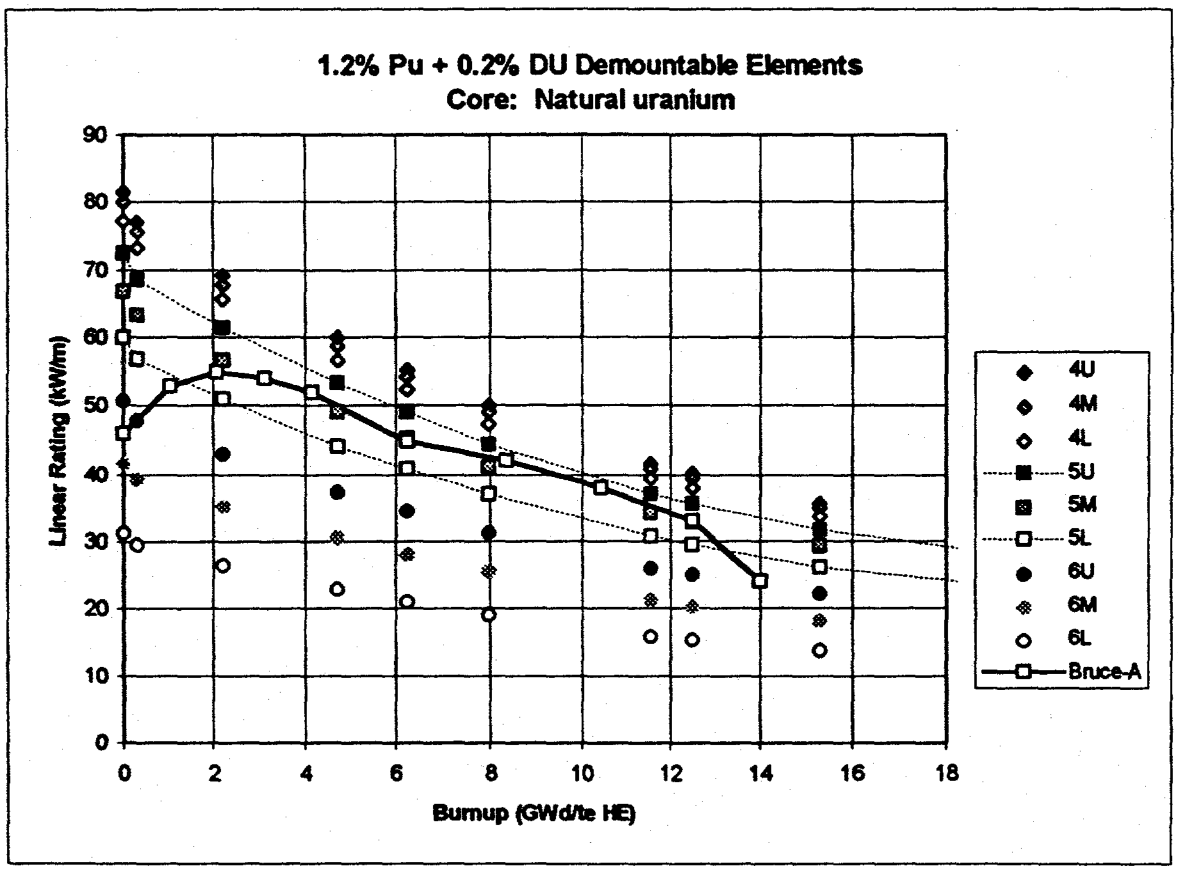

Figure 3: Linear power vs burnup as a function of NRU string position (U-upper, M-middle, L-lower elevations for positions 4, 5 and 6) for 1.2% Pu elements with an NU demountable bundle core. The dashed lines show the upper and lower power limits of a bundle in position 5.

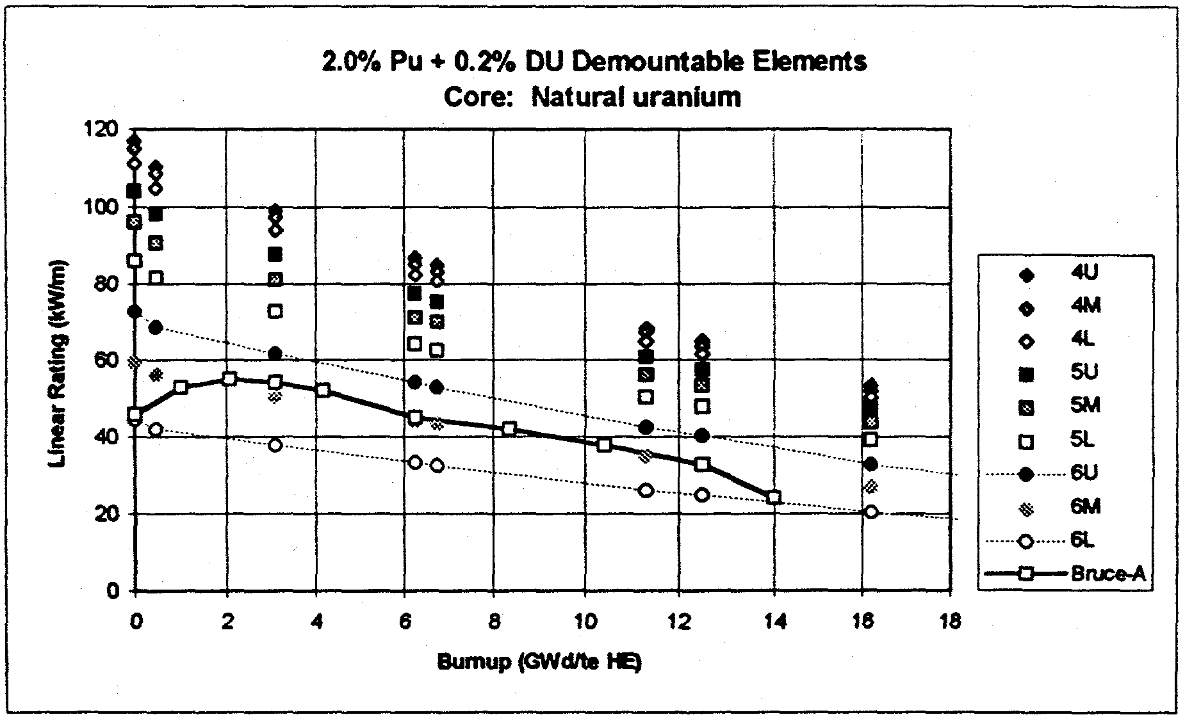

Figure 4: Linear power vs burnup as a function of NRU string position (U-upper, M-middle, L-lower elevations for positions 4, 5 and 6) for 2% Pu elements with an NU demountable bundle core. The dashed lines show the boundary power limits of a bundle in position 6.

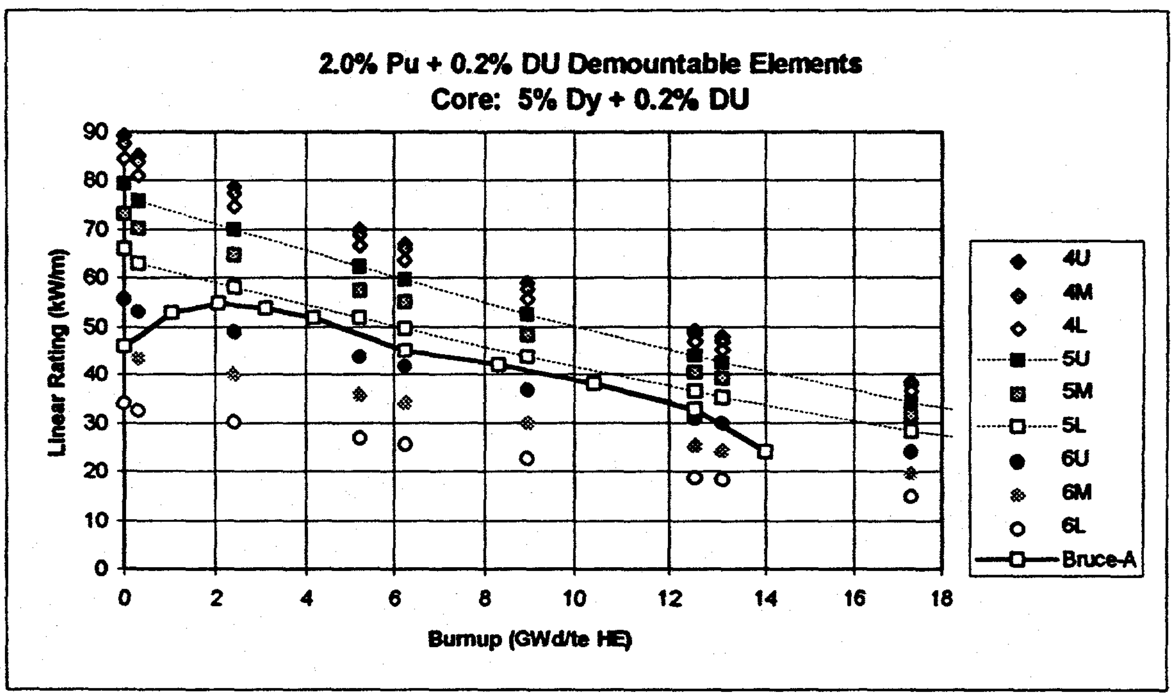

Figure 5: Linear power vs burnup as a function of NRU string position (U-upper, M-middle, L-lower elevations for positions 4, 5 and 6) for 2% Pu elements with a 5% Dy-poisoned demountable bundle core. The dashed lines show the upper and lower power limits of a bundle in position 5.

Each of the four demountable fuel bundles will contain two slightly-enriched UO_2 fuel elements as controls to provide a reference for comparing MOX fuel performance against UO_2 behaviour. The enrichments will be determined by physics analyses, to match the linear powers of the MOX elements at early burnup. These control elements will be located on opposite sides of the outer ring. In addition to these control elements, the Russian and U.S. MOX elements will alternate around the outer ring. Thus, each bundle will contain 2 controls, 8 Russian MOX elements and 8 U.S. MOX elements, as shown in Figure 6.

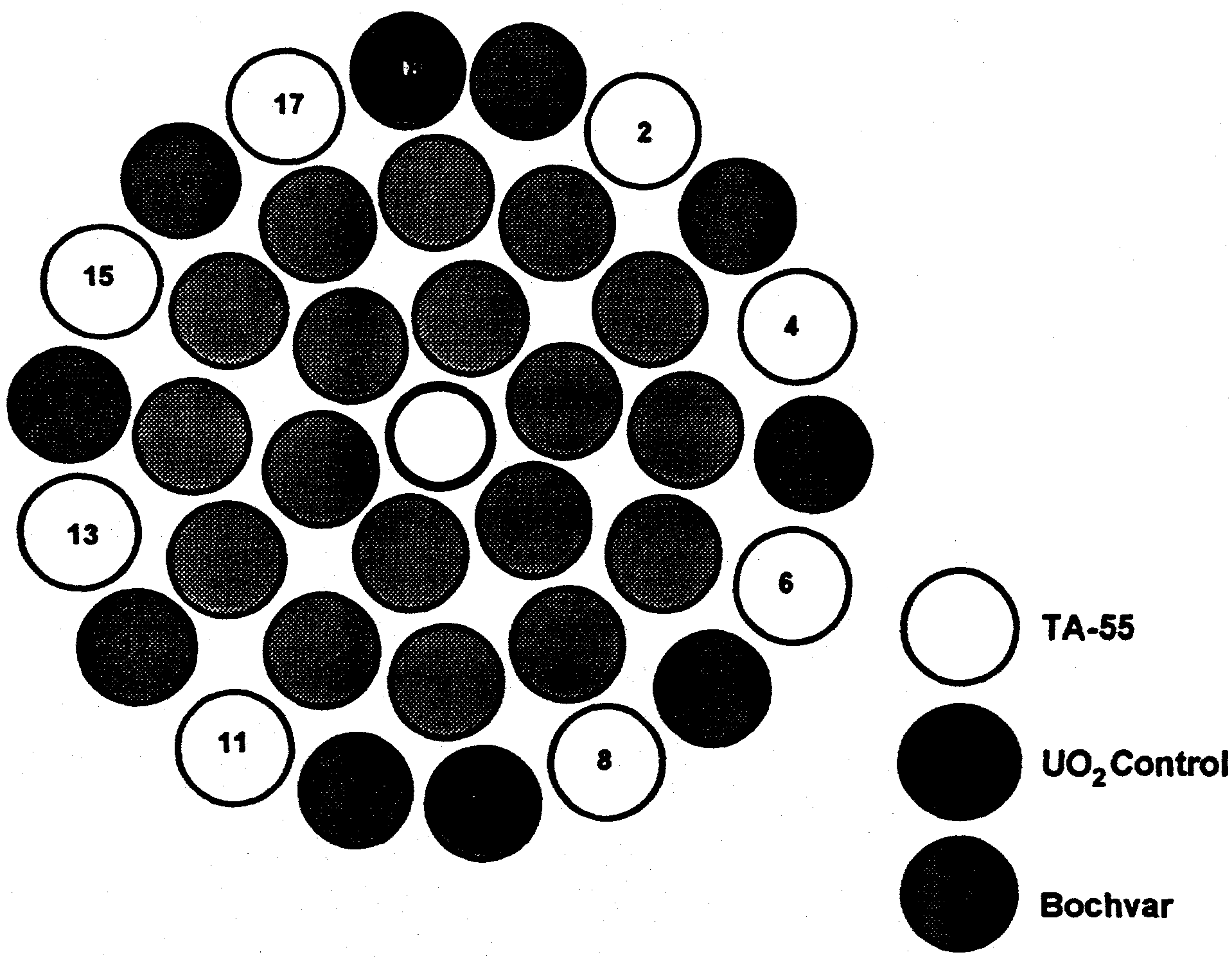

Figure 6: Cross-section of demountable MOX fuel bundle. Two control elements are shown at positions 9 and 18, and the remainder of the outer ring is composed of an alternating mixture of 8 Russian and 8 U.S.-origin MOX elements. The inner two rings (the bundle core) would contain either natural UO_2 (in three of the four bundles), and 0.3% depleted UO_2 with 5% Dy neutron absorber in the other one bundle.

The irradiation test parameters are summarized below in Table 1. The test parameters include three ranges of linear power, two levels of Pu homogeneity, and two Pu concentrations in the MOX fuel, for a total of 7 parameters. The 2% Pu MOX fuel cannot be tested at low linear power in NRU, and thus the total test matrix contains 10 unique sets of test conditions (6 for 1.2% Pu MOX fuel and 4 for the 2% Pu MOX) as shown in Table 1.

Table 1 also indicates the position in the NRU loop that will be used to obtain the ratings for each fuel composition. There are six axial positions available in the vertical fuel irradiation loops, with cosine flux profile along the six positions, such that positions 1 and 6 (top and bottom) are low flux sites in the outer region of the core, positions 2 and 5 are intermediate flux sites, and positions 3 and 4 are neighboring high flux sites centered around mid-elevation of the core. A Dy-poisoned demountable bundle core will be required for the high-powered 2% Pu MOX fuel (sets 9 and 10) in bundle #4. The other three bundles (#1, #2 and #3) will be natural UO_2 (NU) cores.

Table 1: Summary of Parallex Irradiation Test Parameters

Test Parameter		set 1	set 2	set 3	set 4	set 5	set 6	set 7	set 8	set 9	set 10
Linear Power (kW/m)	**35-40**	*			*						
	50		*			*		*	*		
	65			*			*			*	*
Plutonium Homogeneity	**intermed.**	*	*	*				*		*	
	high				*	*	*		*		*
Plutonium Content	**1.2%**	*	*	*	*	*	*				
	2.0%							*	*	*	*
Loop Position		1/6	2/5	3/4	1/6	2/5	3/4	1/6	1/6	2/5	2/5
Bundle Core Type		NU	NU	NU	NU	NU	NU	NU	NU	Dy	Dy
Demountable Bundle #		1	2	3	1	2	3	1	1	4	4

Comparison of data from the various sets identified in Table 1 will provide data on fuel performance as follows:

Parameter sets 1, 2 and 3:	- linear power for 1.2% Pu MOX, intermediate homogeneity
Parameter sets 4, 5 and 6:	- linear power for 1.2% Pu MOX, high homogeneity
Parameter sets 7 and 9:	- linear power for 2% Pu MOX, intermediate homogeneity
Parameter sets 8 and 10:	- linear power for 2% Pu MOX, high homogeneity
Parameter sets 1 and 4:	- Pu homogeneity in 1.2% Pu MOX, low rating
Parameter sets 2 and 5:	- Pu homogeneity in 1.2% Pu MOX, intermediate rating
Parameter sets 3 and 6:	- Pu homogeneity in 1.2% Pu MOX, high rating
Parameter sets 7 and 8:	- Pu homogeneity in 2% Pu MOX, intermediate rating
Parameter sets 9 and 10:	- Pu homogeneity in 2% Pu MOX, high rating
Parameter sets 2 and 7:	- Pu content, intermediate Pu homogeneity, intermediate rating
Parameter sets 6 and 10:	- Pu content, high Pu homogeneity, high rating

In addition, comparisons can be made between US and RF MOX fuel within every parameter set, giving information on the effect of differences in impurity concentrations and any differences in MOX fabrication methods.

There are three target burnups (5, 10 and 15 GWd/te HE), but only a limited number of elements will be removed at the intermediate burnups. Figure 7 depicts the composition of the outer ring of each of the four bundles during the entire irradiation period, and illustrates the sequence of element replacements at various burnup intervals. The vertical black bars in Figure 7 indicate when an element is removed from the bundle.

Bundle #1 will begin irradiation at the earliest date, following shipment of the first batch of MOX fuel from both laboratory fabricators. It will contain only 1.2% Pu MOX of intermediate homogeneity for the first irradiation period (5 GWd/te HE), after which 14 of the 16 MOX fuel elements will be replaced with other compositions, such that this bundle would contain samples of all 4 MOX fabrication variants. The introduction of the other MOX variants is planned after the second shipment of MOX has been processed at CRL.

Bundles #2, #3 and #4 all begin irradiation after the second shipment of MOX. These bundles each contain only two MOX variants.

The irradiation matrix in Figure 7 requires the following MOX fuel to be produced by each fabrication laboratory:

Table 2: Quantity of CANDU MOX Fuel Elements Required From Each Fabrication Laboratory for the Parallex Project

MOX Fuel Type	Number of Elements	Archives/ Spares	Total	kg MOX
1.2% Pu, Interm. homogeneity	23	5	28	14
1.2% Pu, High homogeneity	15	5	20	10
2% Pu, Interm. homogeneity	9	4	13	7.5
2% Pu, High homogeneity	9	4	13	7.5
Total Required (each fabricator)	56	18	74	37

In addition to the MOX fuel elements listed above, about 20 slightly-enriched elements will be required as control elements (14 + 6 spares).

An irradiation proposal will be prepared and submitted for review and approval by the NRU Irradiations Review Committee. This will require supporting physics analyses to confirm powers. A fuel string assembly proposal will be prepared, requiring physics and thermalhydraulics analyses to determine string powers and dryout margins in the loop. The string will be assembled under water in the NRU spent fuel bays, and installed in the reactor during a shutdown. The string will remain under irradiation until target burnups are achieved, or string removal is required to accommodate replacement of other bundles, or to accommodate operational or program constraints. Demountable elements will be replaced at required intervals when the string is removed from the reactor. Throughout the irradiation, data will be acquired on loop powers (by calorimetry) and reactor conditions (core power and neutron fluxes measured at sites adjacent to the loops), and fuel bundle powers will be calculated and documented for each irradiation period.

Visual inspections will be conducted during all handling operations in the bays. At the completion of irradiation, elements will be cooled in the spent fuel bays for about 3 months before commencing PIE.

4.5 Post-Irradiation Examination

A detailed PIE plan will be prepared and reviewed by fuel experts at the completion of irradiations and prior to the start of destructive examinations. Since batches of fuel elements will become available at different times following the discharge and cooling of bundles, the PIE will be conducted in batches. The PIE data from each MOX bundle will be documented separately and an overall PIE report will be prepared to summarize cross-comparisons.

An important output from the PIE tasks will be a summary of the Parallex MOX fuel performance as a function of the NRU irradiation conditions, and conclusions drawn about performance that would be expected under conditions in the Bruce-A reactors.

All of the PIE data will be compared to pre-irradiation data for the archive samples of MOX fuel to quantify changes during irradiation, and to assess if the changes are within expected ranges based on prior knowledge of CANDU fuel performance. The data on irradiation conditions will be used to perform computer code fuel performance simulations to predict measured parameters such as gas release, strain, void volumes, and grain size.

4.5.1 Visual Examination and Profilometry

The individual fuel elements will be transferred from the NRU rod bays to the Universal Cells in Building 234. Each element will be examined visually along its length using the stereo microscope. Photographs and videotape will be taken as required. Element diameters and profiles will be measured over the full length of the element at three orientations using a two-transducer LVDT measuring system. The profilometer measures diameters and profiles (bow) to an overall accuracy of 0.01 mm, with a precision of ± 0.004 mm at (2σ).

4.5.2 Axial Gamma Scans

Five fuel elements from each bundle will be axially scanned for gross gamma activity and isotopic activities (Cs-137, Nb-95, Eu-154, Zr-95, Ru/Rh-106) at 1.0 mm intervals to check for flux peaking and pellet-to pellet gaps. The fuel element is rotated at 1 rps while the gamma spectrum is being collected.

4.5.3 Element Puncture and Fission Gas Analysis

Ten elements from each bundle will be punctured to determine gas volumes and end of life internal gas pressures. The whole fuel element is sealed into the puncture tool chamber, and the prepuncture gas rack system volume is determined barometrically. The element is punctured, the gas collected and the system pressure is measured when equilibrium is attained. The postpuncture system volume, which includes the fuel element, is measured and the internal element void volume is determined as the difference between the postpuncture and prepunctured system volumes. End of life internal gas pressures are calculated at standard temperature and pressure (STP).

A one litre gas sample, from five of the ten elements punctured, will be analyzed by mass spectrometry to measure the fission gas composition and the Xe and Kr isotopic composition.

4.5.4 Ceramographic and Metallographic Examination

Samples will be cut from the mid-axial plane of ten fuel elements from each bundle for optical microscope examination of the fuel and sheath. Samples will also be cut from the reference and non-reference end of five elements for examination.

Oxide layer thickness on the inner and outer surfaces of the sheath and CANLUB layer retention will be measured in the as-polished condition. The fuel will be chemically etched to examine grain growth, porosity and the fuel/sheath interface. Grain size and porosity will be measured at the outer pellet edge, the mid-radius and the central region of the pellet. The final stage will be to etch the sheathing to examine the microstructure and hydride distribution. Macrographs and micrographs will be taken as required.

The distribution of heavy elements and fission products will be examined on samples from five elements with the use of alpha/beta autoradiography.

4.5.5 Burnup Analysis

Twenty-five millimeter long samples will be cut from the peak flux position in five elements for burnup measurement. The fuel is chemically dissolved and the burnup is determined by high performance liquid chromatography (HPLC), which uses La-139 as a fission monitor. The need for isotopic analysis is avoided because La-139 is monoisotopic in fission.

4.5.6 Sheathing Hydrogen Analysis

Slices of fuel element, 5 mm wide, will be cut from five elements for the analysis of hydrogen content in the sheathing. The fuel is removed using a combination of mechanical and chemical methods. The hydrogen content of each sample is determined by hot vacuum extraction mass spectrometry (HVEMS).

4.5.7 Scanning Electron Microscope (SEM) Examinations

Fuel cross-sections from five elements will be examined in the SEM to check for Pu homogeneity in the fuel matrix. The SEM is equipped with energy dispersive and wavelength dispersive X-ray spectrometers (EDX and WDX) for elemental analysis, which is useful for studying the distribution of heavy elements and fission products. Image analysis will be used to quantify the Pu homogeneity in the irradiated fuel.

4.5.8 O/M and Microdensity Measurement

Thin cross-sections of fuel from two elements will be cut into small samples (1mm x 1mm x 2mm) at several radial positions using a numerically controlled saw. These samples will have their density measured using a 3-weight immersion technique with a high precision micro-balance, to yield information on densification and swelling. O/M ratios will be measured on the samples using coulometric titration.

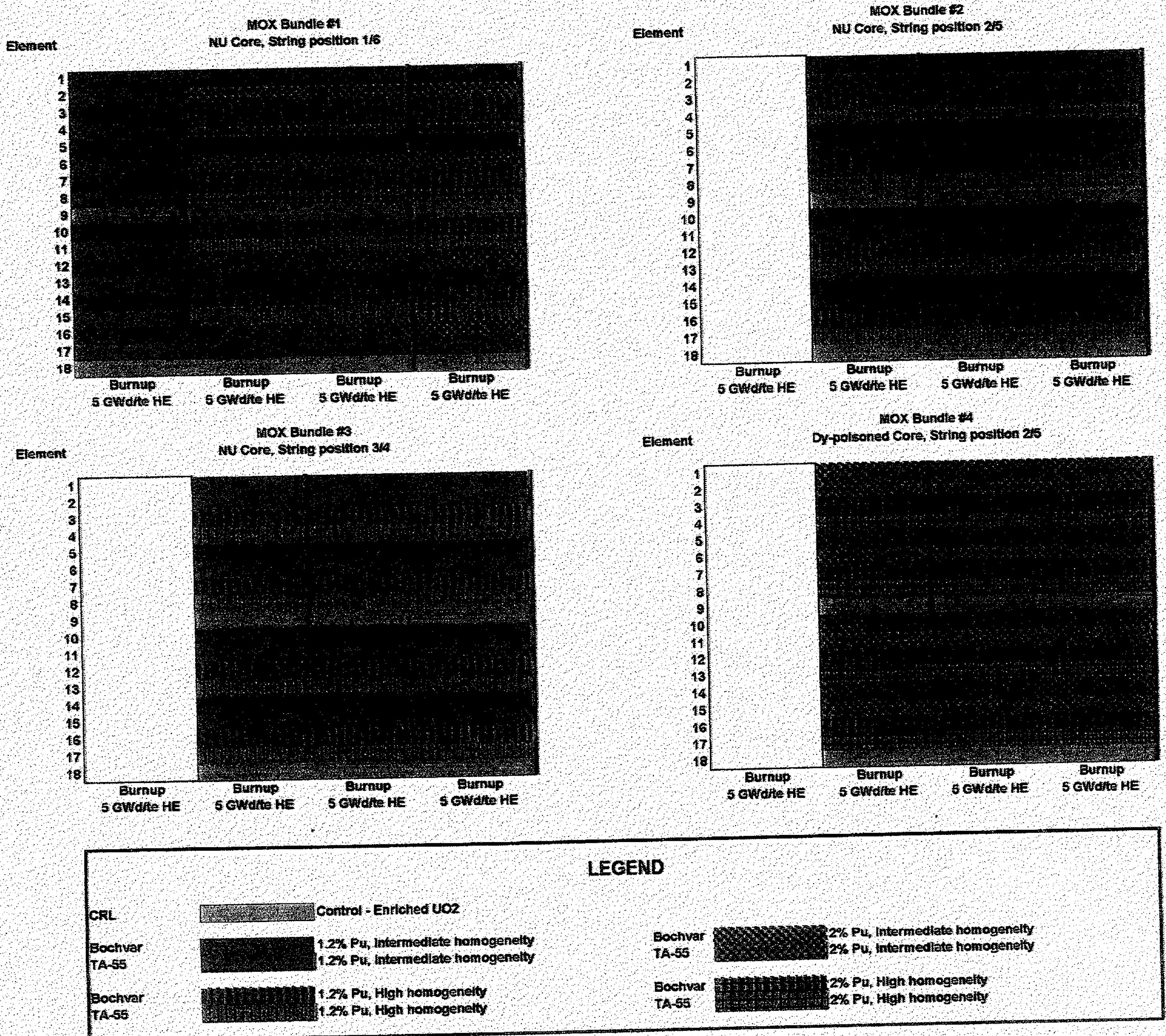

Figure 7: Composition of the 18 elements in the outer ring of the four demountable MOX bundles. The horizontal scale is partitioned into burnup increments of 5 GWd/te HE. Vertical black bars indicate removal of the element from the bundle for PIE.

5. SUMMARY

This document describes the technical basis for the Parallex Project. Further details of the project will be developed once the program is confirmed and is started. These further details will be contained in documents to be prepared that will describe fabrication processes, manufacturing, inspection and test plans, fuel characterization plans, the experimental irradiation proposal, and a post-irradiation examination plan.

The general information in this document provides:

- the technical objectives and rationale for the choice of MOX fuel fabrication parameters that are to be investigated,
- the pre-irradiation fuel characterization plan,
- the NRU irradiation plan,
- the post-irradiation examination plan, and
- a summary of the evaluations that can be extracted from the Parallex data.

6. ACKNOWLEDGMENTS

Physics calculations of fuel element ratings in the demountable bundles were done by M. Atfield. Helpful review comments from H. Feinroth, I. Oldaker, P. Boczar, H. Hamilton and V. Georgiades are acknowledged.

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