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FY2001 Final Report Laboratory Directed Research and Development (LDRD) on Advanced Nuclear Fuel Design in the Future Nuclear Energy Market

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U.S. Department of Energy

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This work was performed under the auspices of the U. S. Department of Energy by the University of California, Lawrence Livermore National Laboratory under Contract No. W-7405-Eng-48.

FY2001 FINAL REPORT

LABORATORY DIRECTED RESEARCH AND DEVELOPMENT (LDRD) ON ADVANCED NUCLEAR FUEL DESIGN IN THE FUTURE NUCLEAR ENERGY MARKET

By

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1. EXECUTIVE SUMMARY

This document examines the business of nuclear fuels in context with future demand and then explores the suitability of using LLNL's computational material science capability and existing ceramification facility for advanced nuclear fuel research and fabrication. The advanced fuels include those different from the standard UO_2 fuel used in current LWRs, and encompass the mixed oxide, metal, nitride, coated particle, and inert matrix fuel. The advanced fuels are designed for (but not limited to) the following purposes:

- Plutonium utilization—mixed oxide fuel
- Fast reactor use—metal or nitride fuel
- High burn-up—coated particle fuel
- Plutonium annihilation—inert matrix fuel
- Thorium based fuel utilization—mixed oxide fuel

The examination indicates that a relatively modest modification to existing ceramification glovebox and equipment in LLNL/B-332 will provide the capability for research and development of inert matrix fuel for plutonium annihilation. Annihilation or "deep-burn" of plutonium is of great interest for non-proliferation purposes and for the disposition of weapons plutonium.

To examine material and manufacturing issues associated with the production of advanced fuels, the ceramification gloveboxline could be converted to a process capable of producing advanced MOX-type fuel. The conversion would include upgrading the operating specifications of the feed preparation equipment and the hydraulic press, replacing the sintering furnace, and adding a centerless grinder. A cost analysis of such modification to the existing equipment was performed, and results indicated a need for an investment of $11.4 M over three years. This includes staffing and equipping the operation and performing initial experimental design.

The advanced fuels and their associated fuel cycles (which consist of the front-end manufacturing, use in the reactors, and the back-end processing and disposal) were evaluated from the non-proliferation and waste management perspectives. To the extent possible, quantitative measures relevant to non-proliferation and waste management are suggested for the comparison of advanced fuel and fuel cycles. It is shown that deeper burn fuels can be designed so as to improve proliferation resistance of spent nuclear fuel. Advanced material science computations and experimentation will be needed to verify if advanced cladding and containment materials are achievable.

To simulate the performance of the advanced fuel in reactor core, we propose to perform core configuration and fuel burn-up calculations with MOCUP, and materials evolution analysis with MDCASK and BIGMAC. These codes are operating in LLNL's high performance computing system enhancing our nuclear fuel/reactor design and materials design and evaluation capability. They represent the opportunity to perform "virtual" fuel and core design, thereby avoiding much of the cost associated with new fuel development and testing programs.

2. HIGHLIGHTS OF ACCOMPLISHMENTS

This project was completed in a period of two months (August and September, 2001) and for a budget of 27K. Highlights of the accomplishments are listed as follow:

- Compilation of available information on advanced fuels, their compositions and designs for various reactor coolants, in previous, existing and future reactors and concepts.
- Evaluation of LLNL's ceramification process for modification to manufacture advanced fuel pellets (MOX), and inert matrix fuel for plutonium annihilation.
- Completion of a cost analysis of modifying LLNL's existing ceramics fabrication capability for advanced fuel fabrication.
- Acquisition of computational and analysis codes and methods, enhancing LLNL's nuclear fuel/reactor design and material analysis capability
- Evaluation of the advanced fuels and their associated fuel cycle (which consists of the front-end manufacturing, reactors, and the back-end processing and disposal) from the non-proliferation and waste management perspectives.

3. INTRODUCTION

This study is to research the maturity of advanced nuclear fuel and cladding technology and to explore the suitability of existing technology for addressing the emerging requirements for Generation IV reactors and emerging thermal/fast spectrum reactors, while simultaneously addressing nuclear waste management, and proliferation resistance concerns.

3.1 Background and Motivation

Nuclear Power currently makes up ~20% of the United States electrical power base and ~16% of the international electrical power base. As the United States seeks a sustainable 21st century energy infrastructure that reduces reliance on Middle Eastern oil and minimizes both greenhouse gas emissions and environmental damage, nuclear power is a primary technology capable of providing a clear path forward. In addition, the worldwide demand for electrical power is projected to increase between 20% and 50% over the next 20 years, with a large number of developing countries seeking to add electrical capacity to improve quality of life. It is forecast that nuclear power will be a major contributor internationally in providing for this demand.

A number of issues, including the management of nuclear waste, concerns over the proliferation of weapon useable fissile materials, safety, and cost remain to be resolved before a significant expansion of nuclear energy occurs. Notably, the development of advanced nuclear fuels can address aspects of all these issues, including the waste management issue, through the production of higher burn-up fuels that minimize the waste stream. Clearly, changes in fuel design, fuel utilization, and spent fuel handling are essential if nuclear power is to be a participant in providing electrical power while not insulting the environment. Advanced reactor fuels must be designed, taking the entire reactor and power production system into account.

This LDRD Project describes the development of a comprehensive advanced nuclear fuels R&D capability. It begins with a compilation of previous and existing fuel fabrication capabilities, as well as recent technical advances in new fuel developments. This will include mixed oxide, nitride, coated particle, cermet, and metal fuels together with an assessment of capabilities to fabricate test fuels. These capabilities are compared for different advanced fuel technologies in various reactor configurations. Configurations include fast spectrum, gas cooled, lead-bismuth cooled, and small modular pebble bed.

In addition, the modern advanced ceramic fabrication capability existing within Building 332 at LLNL has been assessed in addressing the fuel synthesis and cladding design requirements. A cost analysis for modifying the existing equipment, activating and operating the modified equipment was prepared.

3.2 Grand Vision

The grand vision is to demonstrate that advanced reactors and fuels can be designed so as to address proliferation and safety concerns, and reduce some of the legacy associated with lingering spent nuclear fuel.

This is accomplished by evaluating fuel composition and synthesis, cladding design, and design of other associated materials requiring high durability.

To achieve this vision, it is important to pursue the:

(1) Development of a virtual lead test assembly (VLTA) computational capability utilizing LLNL's world-premier high performance, parallel computing infrastructure provided by the Accelerated Strategic Computing Initiative (ASCI), and a package of state-of-the-art computer codes in neutronics, fuel burnup, and materials performance, and

(2) Development of a state-of-the-art fuel fabrication facility capable of manufacturing advanced nuclear fuel test pellets.

The VLTA is a prime example of the new materials science paradigm for computational design of advanced materials. The fuel fabrication facility at LLNL capitalizes on the Laboratory's ability in manufacturing involving a significant quantity of actinides in an integrated R&D facility.

3.3 Project Overview

In the subsequent sections of this document, we first provide (in Section 4.1) a compilation of available information on advanced fuel designs involving mixed-oxide, metal, nitride, coated particle, and inert-matrix (cermet and pyrochlore), in different types of reactor coolants. Next we describe (in Section 4.2) the computational tools acquired for the development and implementation of a virtual lead test assembly (VLTA) to evaluate advanced nuclear fuel and cladding designs. The evaluation of advanced nuclear fuel, included in Section 5.0 encompasses their technical properties (such as chemical, thermal, neutronics and mechanical) and manufacturing techniques and methods, as well as institutional considerations, such as non-proliferation and waste management.

We also provide (in Section 6.0) results of a scoping study of the modification, cost and schedule for converting and activating the Plutonium Ceramification Test Facility (PuCTF) for an advanced fuel fabrication application.

4. Nuclear Fuel Characteristics – Current And Future Reactors

This section describes the advanced nuclear fuels in various reactor coolants. In this document, we define advanced nuclear fuels as those different from the standard UO_2 fuel used in current LWRs, and encompass the mixed oxide, metal, nitride, coated particle, and inert matrix fuel. The advanced fuels are designed for (but not limited to) the following purposes:

- Plutonium utilization in mixed oxide fuel,
- Fast reactor use, such as metal or nitride fuel,
- High burn-up, such as coated particle fuel,
- Plutonium annihilation, as in inert matrix fuel
- Thorium-based fuel utilization, such as mixed oxide fuel

4.1 Advanced Nuclear Fuels in Various Reactor Coolants

Table 4.1 lists a matrix of information on nuclear fuel in various coolant/reactor environments. The Table presents information on advanced nuclear fuel in existing or past reactors, as well as concepts for future reactors.

Table 4.1 Advanced Nuclear Fuel in Various Coolants

Fuel /Coolant	**Mixed-Oxide**[1]	**Metal**	**Nitride**[2]	**Coated Particle**[3]	**Others** (Inert Matrix[4], Molten Salt[5])
H_2O:					
H_2O moderated	LWR/MOX[6]				*Erbium Cermet*[9], *ROX*[10]
D_2O moderated	*CANDU/MOX*[7]				
Gas:					
He				Oxide fuel: Fort St.Vrain, Peach Bottom-1, HTR-10[8], *PBMR*[11], *MHTGR*[12], *Thorium fuel*	
CO_2		Magnox			
Liquid metal:					
Na	Phenix, JOYO, MONJU, FFTF, BR-10, BN-600	EBR-II, *ALMR*[13], *4S*[14]	BR-10		
Pb/Pb-Bi		*4S*	Russian Naval Rx, *Brest*[15], *ENHS*[16]		
Others:					
Li			*SP-100*[17]		
Molten Salt					MSRE[5]

Italic – Conceptual design
Notes:

1. UO_2-PuO_2
2. Mono-nitride
3. Such as TRISO particle
4. Inert Matrix Fuel (Yttria stabilized Zirconia)
5. Such as the Molten Salt Reactor Experiment (MSRE) operated by ORNL in the late 70's
6. Light water reactors (Pressurized and boiling water reactors) fueled with MOX
7. Canadian Deuterium Natural Uranium Reactor fueled with MOX
8. 10 MWe high temperature reactor operated by INET, Tsinghua University, China
9. Erbium Cermet fuels (Oxide fuel doped with Erbium)
10. Rock-like Oxide Fuel
11. Pebble Bed Modular Reactor
12. Modular High Temperature Gas-cooled Reactor
13. Advanced Liquid Metal Fast Reactor (such as the Integral Fast Reactor, designed by ANL)
14. Super Safe, Small and Simple Liquid-Metal Reactor
15. A Russian design Liquid-Metal Reactor
16. Encapsulated Nuclear Heat Source
17. GE-designed Reactor for space application

4.1.1 H_2O Coolant

H_2O Moderated: LWR/MOX

The commercial application of MOX fuel in H_2O cooled LWRs was started in the mid 1980s. Currently, the use of MOX fuel has been established on an industrial scale in a number of countries. In Belgium, France, Germany, Japan and Switzerland, a considerable number of thermal power reactors (PWRs and BWRs) are either licensed (i.e. 40 licensed reactors of which 32 have MOX fuel loaded) or have applied for a license (about 13) to use MOX fuel at levels of up to 30% of the reactor core (see Table 4.2).

Table 4.2 Status of large scale MOX fuel utilization in thermal reactors[1], as of year-end 1998

	Number of Thermal Reactors			
Country	**Operating**	**Licensed to use MOX Fuel**	**Loaded with MOX Fuel**	**Applied for MOX License**
Belgium	7	2	2	
France	57	20	17	8[2]
Germany	19	12	10	4
Japan	52	3	0	1
Switzerland	5	3	3	
Total	130	40	32	13

Notes:

1. There are a number of reactors, notably in Europe and India, not included in this Table, which are licensed to use MOX on an experimental basis,
2. Technically capable reactors planned to be licensed.

MOX fuels are currently used as replacement fuels in LWRs. They are partially loaded in the reactor core. The MOX fuel assemblies design is basically the same as that of the UO_2 fuel

assemblies to preserve the thermal-mechanical integrity of the reactor. The plutonium contents (total or fissile) and the burn-up for the MOX fuel assemblies are limited such that when they are loaded into the core, they would not compromise the safety margins established as the licensing bases for the reactor.

MOX performance in a reactor has been well evaluated for the safe use in thermal reactors. In the past decades, a particular concern was placed on the fission gas release from MOX fuel that might influence fuel rod safety under irradiation. However, it was proved that the release from MOX fuel fabricated by the advanced technology was not significantly different from UO_2 as far as the present burn-up level is concerned. The other MOX fuel behavior characteristics would be the same as that of UO_2. For increasing reliability on MOX fuel use, further efforts are required to accumulate, particularly focusing on safety related behavior such as reactivity insertion accident (RIA) conditions. The database of post irradiation examination (PIE) characteristics of MOX fuel supporting better understanding of in-reactor behavior are rather concentrated in the region of plutonium concentration higher than 20 wt % which corresponds to FBR fuel. It would be necessary to accumulate the MOX data in the region of lower plutonium content.

H_2O Moderated: LWR/Inert Matrix Fuel (Erbium Cermet and ROX)

Basic research has been conducted aimed at almost complete burning of plutonium in LWRs, making it possible to dispose of the spent fuel directly in a geologic repository. This fuel cycle consists of producing chemically stable inert matrix fuel (IMF) in conventional fuel fabrication facilities, burning such fuels in LWRs and disposing of them without further processing.

Inert matrix fuels are tailor made multi-phase fuels consisting of mineral-like compounds, such as yttria stabilized zirconia (YSZ), magnesium spinel ($MgAl_2O_4$) and corundum (Al_2O_3), etc. These mineral-like (or Rock-like) fuels (ROX) are chemically and thermodynamically so stable that they are not soluble in nitric acids in normal conditions.

The IMF does not contain fertile materials, such as uranium or thorium, and as such the production of fissile isotopes (^{239}Pu or ^{233}U) can be suppressed. The absence of ^{238}U in the IMF has a major impact on the fuel management strategy that is used in plutonium annihilation. As formation of ^{239}Pu is eliminated, the reactivity of the fuel lattice drops rapidly with fuel burn-up. To improve the reactivity coefficients, especially for response to a reactivity insertion accident (RIA) in a LWR, some amount of resonant nuclides (^{238}U or ^{232}Th) can be added to the ROX, or to load ROX assemblies partially in a UO_2 fuel core. This additive is effective to reduce the large burn-up reactivity swing for a ROX–filled LWR.

Cermet (ceramic-metal) fuel is formed by adding Erbium metal to oxide fuel. ^{239}Pu has a resonance fission cross section at neutron energy level around 0.3 eV. Burnable-poison materials, such as Er_2O_3 are needed to compensate for a higher loading of ^{239}Pu and to eliminate the potential positive reactivity effect.

D_2O Moderated: CANDU/MOX

CANDU (Canadian Deuterium-Natural Uranium Reactor) is a heavy-water (D_2O) moderated, light-water (H_2O) cooled reactor fueled by natural uranium dioxide. Refueling is done when reactor is on power.

Two advanced nuclear fuel designs utilizing plutonium in a CANDU reactor were proposed for the DOE ex-weapons plutonium disposition program. One design utilizes a standard CANDU fuel bundle that contains 37 elements. The outer rings of 30 elements contain MOX fuel composed of plutonium and depleted uranium oxide. The inner ring of 7 elements contains a mixture of depleted uranium oxide and a burnable poison, dysprosium oxide, which compensates for the excess reactivity introduced by the plutonium. The fuel would operate within the same burn-up and power rating envelop as standard CANDU fuel and would have similar nuclear parameters that allow the reactor to operate within its existing safety and licensing envelope.

A second design utilizes 43 elements in a new bundle configuration called "CANFLEX. The bundle elements have a smaller diameter which allows the reactor to be operated at a lower linear power rating, and thus would permit higher plutonium concentrations and higher burn-up.

A more complete destruction or annihilation of plutonium in a CANDU reactor fueled with advanced fuel is also possible. The advanced fuel would consist of a mixture of plutonium isotopes in a neutronically inert matrix. The absence of ^{238}U eliminates the source for further production of plutonium.

4.1.2 Gas Coolant

He cooled (MHTGR, PBMR)

In the current design of high temperature gas cooled reactors (HTGRs), either fuel-block Modular High Temperature Gas-cooled Reactor (MHTGR) or Pebble-Bed Modular Reactor (PBMR), the Triso-coated fuel particles are used. The Triso-coated particle consists of a microspherical fuel kernel and coating layers of porous pyrolytic carbon (PyC), inner dense PyC, silicon carbide (SiC), and outer dense PyC. The porous PyC coating layer, called the buffer layer, attenuates fission recoils and provides void volume for gaseous fission products and CO gas. The inner PyC coating layer acts as a containment to the gases. The SiC coating layer provides mechanical strength for the particle and acts as a barrier to the diffusion of metallic fission products. The outer PyC coating layer protects the SiC coating layer mechanically.

Coated particle fuels were manufactured by General Atomics (GA) for Peach Bottom Unit 1 and the Fort St. Vrain. GA is also collaborating with Russia in designing a MHTGR utilizing the Triso-coated particle fuel loaded with Pu-oxide kernel for plutonium annihilation. GA's experience with Pu particle fuel in Peach Bottom 1 reached a burn-up level of ~750GWtD/t.

Currently, the Institute for Nuclear Engineering and Technology (INET) of Tsinghua University in Beijing, China is operating a 10 MWe Pebble-bed Gas cooled Test Reactor to demonstrate the safety features and to acquire the operational experience of electricity generation with a modular HTGR. The INET adopted the AVR (a 300 MWe Pebble-Bed Gas-Cooled Reactor) technology and the know-how in manufacturing the coated-particle fuel from Germany. They manufacture the coated-particle "pebbles" filled with "Triso" particles for the reactor.

A similar concept for a Pebble-Bed Modular Reactor (PBMR) is being promoted by Eskom of South Africa (where a US's utility Exelon is an investment partner). The burn-up of Triso fuel in the PBMRs (both the INET 10MWe reactor and the Eskom design) is around 80-100 GWtD/t.

CO_2 cooled (Magnox)

BNFL of UK operates CO_2 cooled, graphite-moderated reactor using natural uranium metal fuel rods. The fuel rods are contained in casings made of a magnesium alloy called 'magnox'. In a Magnox reactor, the uranium metal fuel reaches temperatures up to 460°C. Carbon dioxide flows around the hot magnox fuel rods and fins and carries away their heat. Refueling of a Magnox reactor is done while the reactor is at power.

Some advanced reactor CO_2 cooled concepts have been developed in Japan. Otherwise CO_2 as coolant is not of much interest.

4.1.3 Liquid Metal Coolant

Sodium-cooled

The sodium cooled fast reactors have been developed using oxide or metal fuel, and primarily stainless steel cladding. Table 4.3 shows the sodium cooled fast reactors operated in various countries, their fuel and cladding materials, the achieved burn-up and current status. Both of these fuels (oxide and metal) have considerable operating, manufacturing and reprocessing experience supporting their use in sodium.

Table 4.3 Sodium Cooled Fast Reactors

Country	Reactor	Rating, MWe	Fuel	Cladding	Burn-up, GWd/t, (%)	Status
France	Phenix SuperPhenix	250 1240	MOX[1] MOX	SS[2] SS	100 70 (1st core)	Operation Shutdown
Japan	Joyo Monju	0 280	MOX MOX	SS SS	75 80	Operation Stand down
Kazakhstan	BN-350	150	UO_2	SS	100	Shutdown
Russia	BR-10 BOR-60 BN-600	-- 12 600	MOX, UN MOX UO_2	SS SS SS	(8.7) (33) 100	Operation Operation Operation
US	FFTF EBR-II	-- 20	MOX U-Pu-Zr Alloy	SS Ferritic-Martiensitic Steel	15 (20)	Stand down Shutdown
UK	Dounrey	--	UO_2	SS		Decommissioning

Notes:
1. UO_2+ PuO_2
2. stainless steel

The cladding material is especially important. It provides not only an interface between the fuel and coolant but also a barrier to prevent fission gas release. Most liquid-metal cooled reactors use stainless steel as cladding to retain fission gases, and satisfy the material compatibility requirement as interface between the metal fuel and liquid-metal coolant.

A small fast reactor design called 4S (stands for Super Safe, Small and Simple) is an innovative liquid-metal cooled-reactor concept having both primary and secondary sodium coolant systems. The core thermal power is 125 MWt and it uses a superheated steam cycle to produce 50 MWe. The reactor uses metal fuel and the coolant is force-circulated using an EM-pump. A unique design feature is that it is tall and slender, and the reactivity is controlled with moving reflectors internal to the reactor vessel. The design is the product of a Japanese (CRIEPE/Toshiba) team. The concept has received considerable interest and support from Japanese utility (Chubu Electric).

Lead/Lead-Bismuth Cooled

The oxide and metal fuels are less likely to be used in Pb-Bi cooled fast reactors. Primary concerns are the low thermal conductivity of oxide fuel, and for metal fuel, the active interaction between the fuel and lead in the case of cladding breach. These concerns could be mitigated by design modification.

A less developed fuel such as, nitride, appears to be the preference for Pb-Bi cooled reactors. The Russian Naval submarines use nitride fuel (UN) in Pb-Bi coolant. Although the operating experience of the Russian submarine is not readily available, such combination of fuel and coolant has been proposed by Russia for commercial-size liquid-metal cooled fast reactors (BREST-300 and BREST-1200).

A version of the 4S fast reactor concept, the Encapsulated Heat Source (ENHS) reactor uses nitride fuel in a lead-bismuth coolant. The primary advantage of using Pb-Bi as coolant is the elimination of a secondary coolant loop and the sodium/water relief system. This simplifies the reactor system and eliminates safety concerns involving liquid metal spills.

4.1.4 Other Coolants

Lithium-cooled

A lithium (Li) cooled reactor using nitride fuel called SP-100 was designed for space application. The reactor is of modular design with a unit power rating of 100 KWe. Unique design features of the SP-100 include long life, high reliability and high survivability in an outer space environment. High-density uranium nitride fuel pins were manufactured. Test fuel was irradiated in Nb-1Zr cladding. Irradiation of 8 test pins indicated no fuel relocation or defect occurred at expected burn-ups. In the reference design, the cladding would be Rh-lined Nb-1Zr.

Molten Salt: MSRE

The molten salt reactor experiment (MSRE) operated by ORNL in the late 70's used a liquid solution of uranium as fuel and removes heat from the reactor by circulating hot fuel to an external heat exchanger. No reactor coolant is employed other than the fuel itself. Neutron moderation was provided by Graphite blocks containing channels through which the molten salt flowed. The MSRE was terminated after a short period of operation.

A molten salt reactor utilizing thorium could be operated as a thermal breeder. The fuel would be a solution of UF_4 in a solvent salt consisting of mixture of BeF_2, ^7LiF, and ThF_4. Separated ^7Li is required instead of the natural lithium because the ^6Li in natural lithium absorb too many neutrons to make breeding impossible. The reactor will breed ^{233}U from neutron capture in thorium.

4.2 Computational Nuclear Fuel Design Tools

The requirements for advance fuel designs have to be justified with detailed analysis of reactor physics, neutronic properties, and material performance. This is accomplished by applying an integrated computer analysis with state-of-the-art computer codes. For this program, we have acquired two sets of such codes to perform neutronic analysis on core configuration design and fuel burn-up in addition to the codes we have to evaluate materials evolution in the irradiation environment.

Neutronic analysis by MOCUP

MOCUP is a package of computer codes coupling MCNP and ORIGEN2. MCNP is a Monte-Carlo transport code with continuous energy groups of neutron cross-sections. It is a state-of-the-art code, specially designed for complex geometry systems. ORIGEN2 is a depletion code, used to calculate the generation and depletion of radioactive isotopes in a reactor system.

MOCUP is a "driver" code that executes recursively between MCNP and ORIGEN2 to perform the particle transport and the desired depletion of a reactor system. MOCUP, serving as "format bridges" between the codes, provides a path to use the MCNP's generalized-geometry Monte-Carlo transport technique and the ORIGEN2's depletion and isotopic generation modules. This allows general materials (fuel, target, cladding, coolant, control rods, etc.) to be depleted in a neutron field, with the accuracy of a transport neutronics solution.

Material evolution evaluation by MDCASK, BIGMAC

MDCASK is a highly optimized molecular dynamics code for parallel computing. The code uses the message passing interface library (MPI). It was originally developed for a CRAY T3D, and later optimized for the ASCI Blue (IBM) machine using OpenMP. It runs efficiently on other platforms, including DEC Alpha (tc2k) and newer IBM (ASCI Frost) machines. MDCASK uses semi-empirical interatomic potentials and has sub-routines to treat systems with potentials based on Stillinger-Weber, Tersoff, embedded atom method (EAM) and modified embedded atom method (MEAM). The MDCASK code tracks the formation of defects in the high-energy displacement cascades which result from the collisions between high-energy particles such as neutrons and fission products and lattice atoms.

BIGMAC is a serial, kinetic Monte Carlo code to simulate the time-evolution of defects and microstructural features. The code employs a 3-dimensional, atomic-scale model, focused on defects, not atomic vibrations. Probabilities for all given events/reactions are obtained from lookup tables. Events are chosen according to their probability. The simulation time is given by the inverse of the total rate of events. BIGMAC has been extensively used to study the accumulation of damage in irradiated materials, the recovery of irradiated materials following isochronal annealing and boron diffusion and activation during semiconductor processing.

The combination of MDCASK and BIGMAC provide a predictive capability to follow the evolution of materials structure during irradiation. The many processes that mediate materials evolution during irradiation are described in the following section. All of these processes are captured in the MDCASK/BIGMAC model.

5. EVALUATION AND RESULTS

5.1 Technical Evaluation

Thermal Properties

Table 5.1 shows the thermal properties (density, melting temperature, heat capacity, thermal conductivity and thermal expansion coefficient) of the various fuel types. These are temperature dependent properties at temperatures of reactor operating conditions. The information provided in the Table is intended for a first-order comparison of thermal properties of these fuel materials. Detailed analyses using these thermal properties are required to evaluate relevant fuel safety criteria related to advanced fuel behaviors. These criteria are:

- Fission gas release and retention
- Fuel pellet cladding interaction
- Helium gas accumulation
- Radiation swelling effects
- Fuel centerline temperature
- Others

In addition, spectrum of operating transients (e.g., control rod ejection or drop) and design basis accidents (e.g., loss of coolant, reactivity insertion accidents) are also required for safety analysis to demonstrate the adequacies of advanced fuels.

Table 5.1 Thermal Properties of Different Fuel Types

	U metal	Pu metal	U+ 10%Zr	UO_2	PuO_2	UC	PuC	UN	PuN
Density, g/cc	19.04	19.84	17.5	10.95	11.44	13.63	13.5	14.31	14.25
Density of Fuel, g/cc	19.04	19.84	15.8	9.7	10.1	13.0	12.9	13.5	13.5
Melting Temp., °C	1134	640	1150	2840	2390	2520	1650	2850	2800
Heat Capacity, J/kg°K(Temp.,°K)	146 (600)	143 (500)	164 (600)	346 (1500)	350 (1500)	160 (700)	165 (700)	251 (1000)	239 (1000)
Thermal Conductivity, W/m°K(Temp.,°K)	36.5 (700)	8.3 (500)	114 (300)	2.3 (1500)	2.2. (1500)	21 (700)	16 (700)	23.7 (1000)	15 (500)

Materials Constraints

A great challenge in the development of new materials for service in advanced reactors is to accommodate the intense neutron irradiation environment, which is known to severely degrade material properties. The attainable burn-up and thus operating life of current nuclear fuels is limited by materials performance issues, which result from changes in the thermal and mechanical properties and dimensional stability of the fuel pellets, cladding and structural steel

during neutron irradiation. The most common material-related failures involve containment breach of the Zirconium-based cladding through a combination of fission gas buildup, pellet clad interaction, corrosion, radiation embrittlement, and thermal and irradiation creep rupture. The development of advanced nuclear fuels will require improved materials and design approaches in order to reach higher burn-ups. As well, the surrounding materials in advanced, high burn up cores will most certainly be required to maintain structural integrity over long service lifetimes in an intense neutron environment, which will severely degrade material properties.

The effect of irradiation on materials microstructure and properties is a classic example of an inherently multiscale phenomenon, involving processes spanning a wide range of length and time scales, as illustrated in Figure 5.1. Changes in material microstructure and microchemistry during irradiation are responsible for changes to and degradation of numerous physical and mechanical properties that impact component (nuclear fuel pellet, cladding, structural support steels) lifetime. At the smallest scales (nanometer and picosecond), high-energy particle (neutrons, gamma-rays, and fission fragments) collisions with lattice atoms result in displacement cascades that produce highly non-equilibrium point defect and point defect cluster concentrations with a high degree of spatial correlation. Concurrently, high concentrations of fission gas products and insoluble helim and reactive hydrogen gas are generated through the fission process and neutron interactions. These primary defects cluster, diffuse and ultimately annihilate over macroscopic length and time scales, significantly altering the chemistry and microstructure of the material, which further impacts the production, clustering and diffusion of subsequent damage at high dose rate and at high dose. Microstructural changes are responsible for degradation of physical properties, including transmittance, reflectance and thermal conductivity; dimensional instabilities, such as swelling and irradiation creep; and mechanical property changes and degradation, such as irradiation hardening and post-yield deformation behavior including plastic flow localization, which impact component performance, reliability and ultimately, lifetime.

Using expertise on dynamic modeling of radiation damage in materials, high performance materials can be designed to survive very high radiation fluxes and retain both thermal characteristics, and containment strength. For example, the deployment of a new high performance cladding would result in significant increases in burn cycle, allowing ^{239} Pu to be significantly burned out of the spent fuel. Fuel would remain in the reactor for longer periods resulting in a higher overall power output, less nuclear waste, and lower costs. Design of high performance cladding alloys, using the knowledge of radiation damage research associated with weapons program and other externally funded efforts is a critical next step.

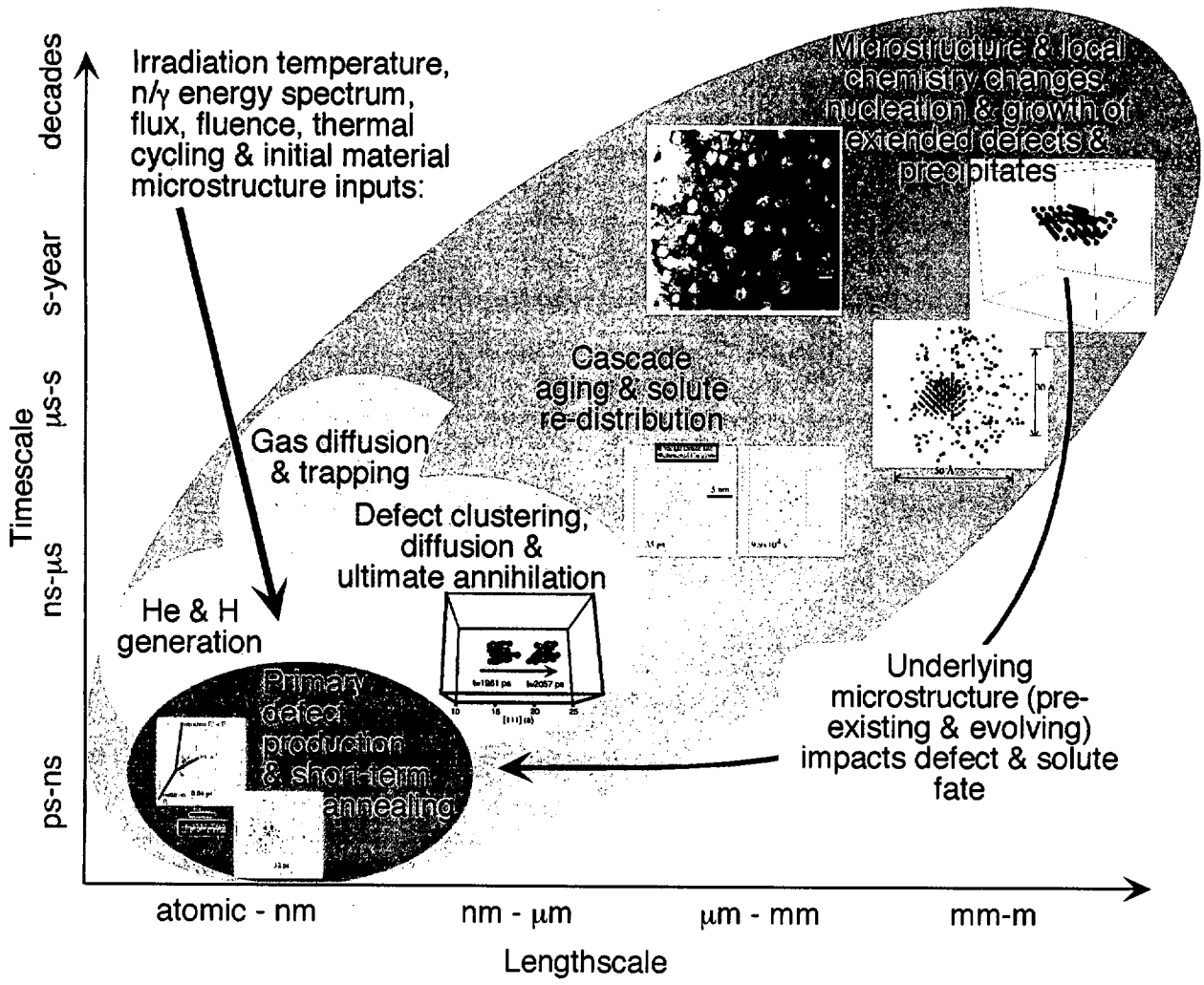


Figure 5.1 – Illustration of the many processes and length and time scales that occur in a material during irradiation in the core of a nuclear reactor. High energy particle irradiation produces displacement damage and insoluble gases over very short timescales (~100 ps) and small volumes. The defects produced subsequently diffuse through the material, resulting in significant clustering and solute re-distribution over much longer time and length scales. The diffusion and clustering of defects produces changes in the material microstructure that impacts (often negatively) the materials properties.

Manufacturing Techniques and Methods

Table 5.2 gives the schematics of the MOX fuel pellet fabrication process used in current MOX fuel fabrication facilities. The process and facilities included are:

- MIMAS at Belgonucleaire's Mol Plant
- COCA at Cogema's CFCa Plant
- SBR at BNFL's MDF Plant
- PNC's PFPF Plant

The process used in MOX fuel fabrication differs from that used in UO_2 fuel fabrication, mainly at the level of the preparation of feed powders. Plutonium is more radiotoxic than uranium. The processing and handling of plutonium require confinement within gloveboxes. The other fabrication steps (mainly granulation, pressing, sintering and grinding) are similar among different MOX fuel processes to steps used in the fabrication of UO_2 fuel.

Table 5.3 shows the schematics of fabricating pellets of nitride fuel, inert matrix fuel, and dirty MOX. LLNL's capability of making ceramic plutonium pucks for the Plutonium Immobilization Program is also included for comparison. Dirty MOX is defined as making a PuO_2-UO_2 mix using process similar to MOX fuel pellet manufacturing but not with the same quality control and specifications required by the fuel manufacturers. The intent for the dirty MOX is to directly dispose of the plutonium in a spent fuel cask containing regular spent nuclear fuel.

Table 5.3 indicates that LLNL's capability in making ceramic plutonium pucks is similar to the capabilities needed for making advance MOX fuels, albeit new equipment such as a centerless grinder and a sintering furnace are needed and operating specifications of existing equipment must be modified. The current LLNL glovebox equipment can be used to make inert matrix fuel as well as dirty MOX with relatively minor adjustments.

Table 5.4 shows the schematics of making coated particle fuel. The process is different from those described in Tables 5.2 and 5.3, and employs different equipment (i.e., droplet formation nozzles/orifices, fluid-beds, special sintering furnace, etc.) to make, coat and fabricate particle fuel into fuel block or fuel pebbles. The Table also shows the schematics of the IFR process of fabricating metal fuel from spent LWR fuel. LLNL's Plutonium Facility (B-332) is capable and has the proper equipment to process nuclear materials in steps similar to many of those employed by IFR.

The comparison of advanced fuel manufacturing techniques and methods as shown in these schematics indicates that LLNL's capability in plutonium ceramification and metal manufacturing is compatible to those employed in industrial and research facilities. Details of the LLNL's ceramification process and equipment, and a cost estimate of modification required for advanced fuel fabrication are included in Section 6.

Table 5.2 MOX Pellet Fabrication Flow Sheets

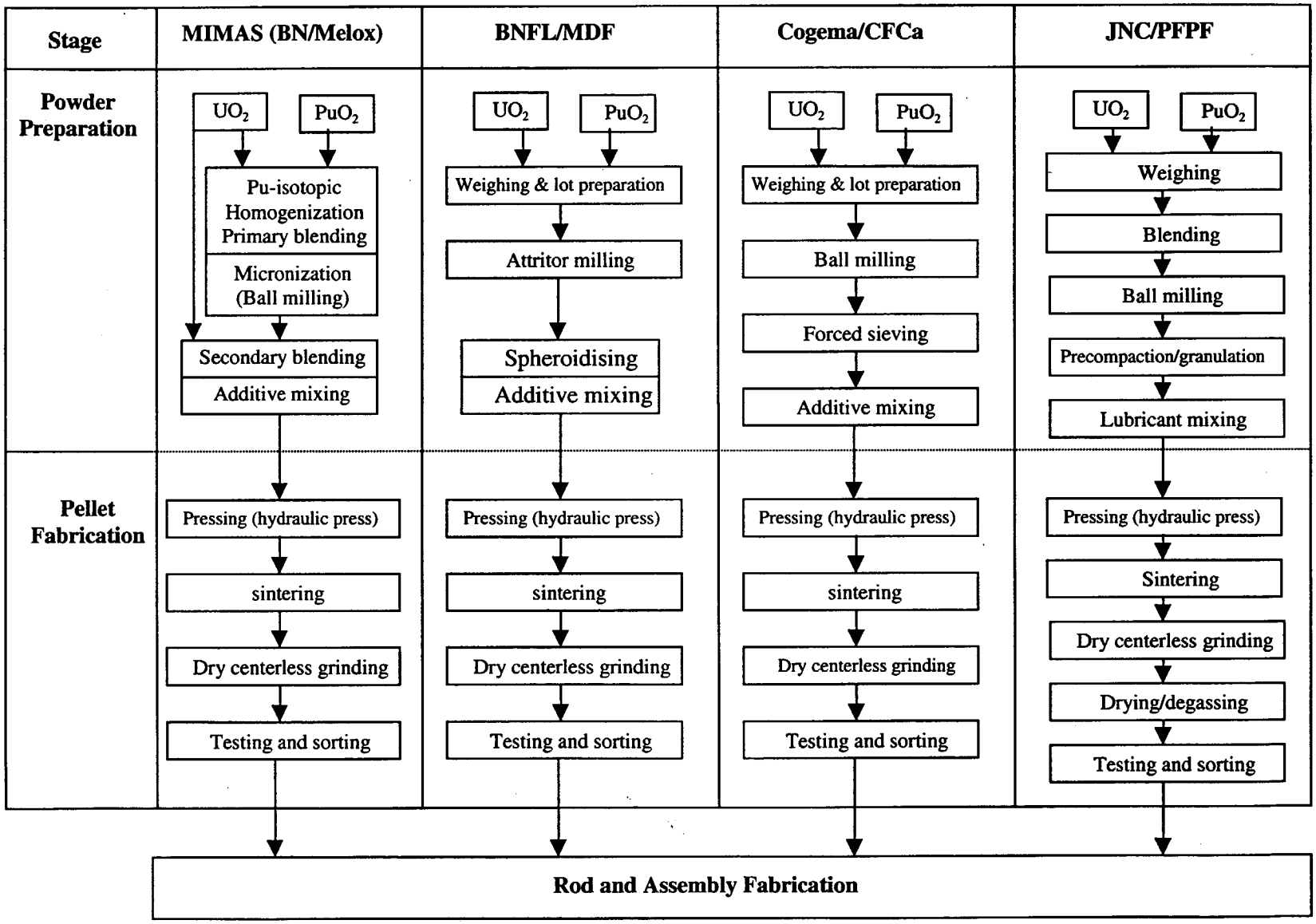
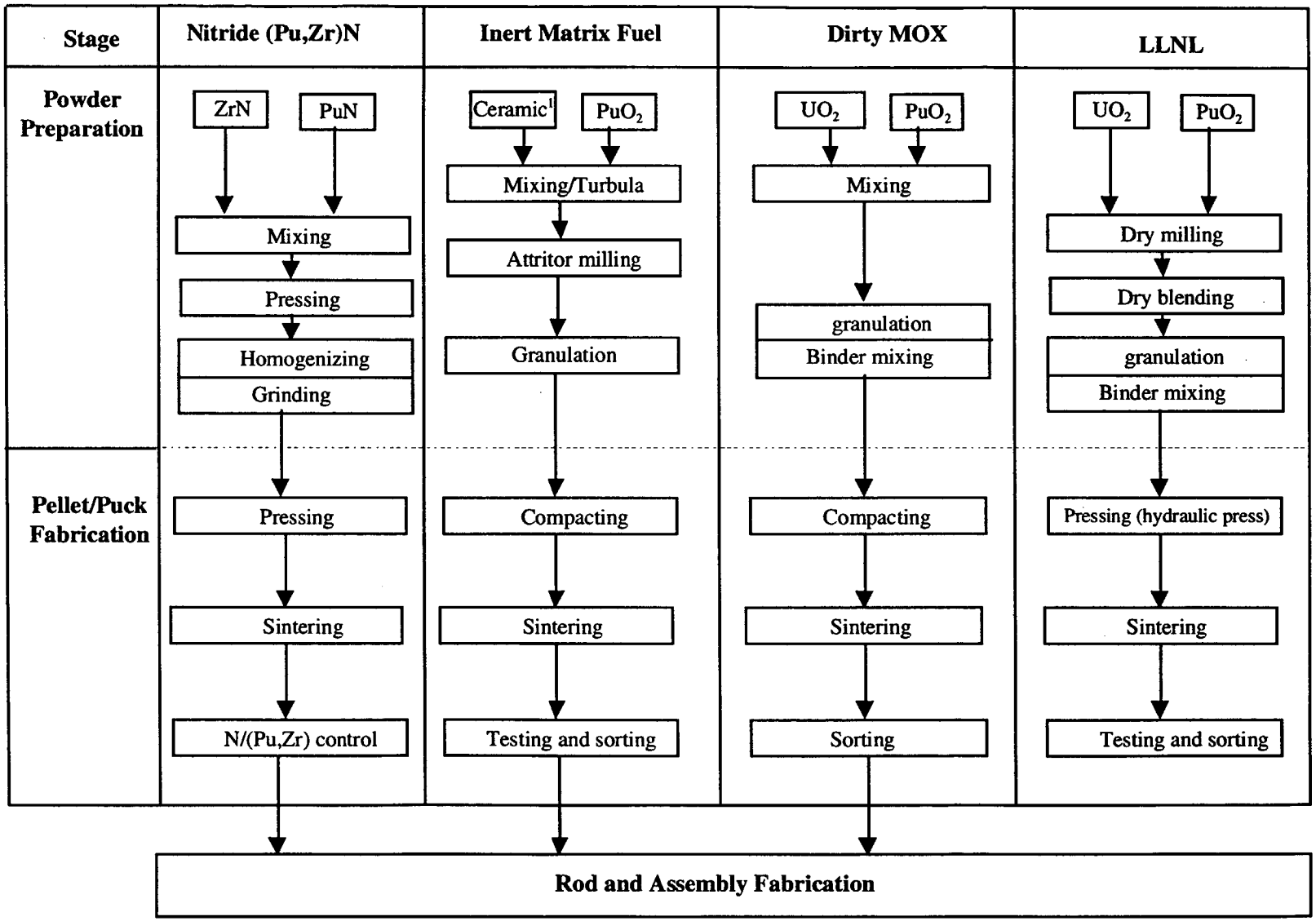


Table 5.3 Advanced Pellet/Puck Fabrication Flow Sheets



1. Includes Er_2O_3, ZrO_2, Y_2O_3

Table 5.4 Metal Fuel and Coated Particle Fuel Fabrication Flow Sheets

Stage	Coated Particle	IFR	LLNL/B-332	
Preparation and Fabrication	Pu metal PuO_2 Nitration Purification Kernel Fabrication Fluidized-Bed/nozzle TRISO Coating Compact Fabrication Element Assembly	LWR spent fuel Disassembly Decladding Direct Oxide Reduction Salt Transport TRU and U separation TRU Electrorefining Cathode Processing Alloying and Slug Casting Pin Fabrication Testing and sorting	Direct Oxide Reduction Motlen Salt Extraction Pu Electrorefining Cathode Processing Alloying and Slug Casting	

5.2 Institutional Evaluation

The advanced fuels and their associated fuel cycle (which consists of the front-end manufacturing, use in reactors, and the back-end processing and disposal) are evaluated from the non-proliferation and waste management perspectives. To the extent possible, quantitative measures relevant to non-proliferation and waste management are suggested for the comparison of advanced fuel and fuel cycles.

Non-Proliferation Considerations

The plutonium quality is a relevant measure for non-proliferation. Degraded plutonium quality decreases its attractiveness for use as nuclear weapons. Civil plutonium separated from reprocessing of high burn-up has a composition very much different from that of weapons grade. The higher content of heat producing isotopes (e.g., ^{238}Pu) and spontaneous fissionable isotopes (^{240}Pu, ^{242}Pu) in reactor-grade plutonium significantly complicate its use as weapons materials. Table 5.5 lists the isotopic compositions of various types of plutonium, from weapons-grade to different reactor-grades.

Table 5.5 Compositions and Ratio of ^{240}Pu to ^{239}Pu for Different Fuels and Fuel Cycles

	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{240}Pu/^{239}Pu
Weapons[1]	0.0001	0.938	0.058	0.0013	0.0002	0.0022	0.062
Reactor[1]	0.013	0.603	0.243	0.056	0.050	0.035	0.403
Magnox[2]	0.002	0.780	0.180	0.034	0.004	--	0.231
LWR[2]	0.015	0.580	0.220	0.135	0.050	--	0.379
VVER-440[3]	0.009	0.610	0.220	0.120	0.041	--	0.361
LWR/MOX[4]	0.035	0.451	0.264	0.160	0.091	0.013	0.585
Candu-NU[5] (8.3 GWd/t)	--	0.641	0.282	0.059	0.018	--	0.440
Candu-MOX[5] (9.7 GWd/t)	--	0.514	0.376	0.086	0.024	--	0.732
Candu-MOX[5] (17.1 GWd/t)	--	0.577	0.345	0.063	0.015	--	0.598
MHTGR[6] (w/W-Pu)	0.018	0.434	0.254	0.195	0.098	--	0.585
MHTGR[6] (deep Pu burn)	0.010	0.069	0.073	0.359	0.489	--	1.058
LMFBR[4], core	0.00078	0.673	0.291	0.028	0.0091	0.0025	0.432
blanket	0.00020	0.952	0.047	0.0013	0.00003	0.0001	0.050
IFR[7] (104 GWd/t)	0.0063	0.747	0.200	0.021	0.0085	0.0069	0.268

Note:

1. "Management and Disposition of Excess Weapons Plutonium (Reactor-Related Options), National Academy Press, Washington DC 1995.
2. "Guidebook on Quality Control of Mixed Oxide and Gd-Bearing Fuels for LWR," IAEA-TECDOC-584, Vienna, Feb. 1991.
3. A. Chesbeskov, Institute of Physics and Power Engineering, Russia, Private Communication, April, 1999.
4. M. Benedict, T. Pigford, H. Levi, Nuclear Chemical Engineering, 2nd. Edition, McGraw-Hill, 1981.
5. Philip Campell, AECL Technologies, Inc., Letter dated 20 September 1994.
6. "MHTGR Plutonium Consumption Study," GA/DOE-051-94, General Atomics, April 29, 1994.
7. T. R. Johnson, et.al., "Use of Transuranic Elements from LWR Fuel in Integral Fast Reactors," ANL-IFR-127, Feb. 1990.

The aim here is to design fuels and fuel cycles which produce plutonium with compositions different from those of weapons-grade (e.g., high ^{238}Pu and high $^{240}Pu/^{239}Pu$ ratio).

Since the dawn of the nuclear era, there has been a concern for misuse of nuclear materials intended for peaceful purposes by its owners, and for theft or diversion of the materials by rogue nations or terrorist groups. The establishment of the International Atomic Energy Agency (IAEA) in 1957 was intended to prevent the potential spread of nuclear-weapons materials and capability. Significant safeguards inspection effort and traditional measures have been and will continue to be used by IAEA. For example, Table 5.6 shows the number of person-days of inspection performed by IAEA annually for typical declared facilities.

Table 5.6 IAEA Safeguards Inspection Effort on Declared Nuclear Facilities

Type of Facility	Person Days of Inspection per Year
Light Water Reactor, no MOX	6 - 12
CANDU Reactor	45
Light Water Reactor with MOX	15 - 45
Enrichment Plant	70 - 150
MOX Fuel Fabrication Facility	~ 200
Reprocessing Plant	> 750

Applying this inspection effort to nuclear fuel cycles employing advanced nuclear fuel, quantifiable measures (in person days of inspection per year) for various reactors and fuel cycles are obtained and compared in Table 5.7.

Table 5.7 Non-Proliferation Measures of Different Nuclear Fuel Cycles

Reactor	**Fuel Type**	**Enrich -ment (70-150 d)**	**Fabrication (MOX Fab: ~200 days)**	**Reactor (Refueling: off-power: 6-12 d, on-power: 45 d)**	**Aqueous Process (>750 d)**	**Pyro./dry Process (>750 d)**	**Inspection Days/year[1]**
Weapon Production	Driver /Production	HEU /No	HEU Fab. /NatU Fab.	**Production Reactor**	Yes /Yes	Yes /Yes	None (exempt)
LWR Once Thru	UO_2	LEU	LEU Fab.	**LWR**	No	No	Rx: 6-12 FC: 70-150
LWR w/MOX	UO_2 /MOX	LEU /No	LEU Fab. /MOX Fab.	**LWR**	Yes /No	No /No	Rx: 6-12 FC: UO_2: 820-900 MOX: ~200
Candu Once Thru	UO_2	No	NatU Fab.	**Candu**	No	No	Rx: 45
Candu /MOX	Canflex	No	Canflex Fab.	***Candu /MOX***	No	No	Rx: 45 FC: Canflex: <200
PBMR, MHTGR	Coated Particle	LEU	LEU Fab.	***PBMR, MHTGR***	No	No	Rx: 45 FC: 70-150
***IFR*, 4S**	LWR: UO_2 /FR: Metal	--- /No	--- /Pu-U-Zr Metal Fab.	***IFR*, 4S**	No /No	Yes /Yes	Rx: 6-12 FC: UO_2: >750 Metal: ~200
Brest, ENHS	LWR: UO_2 /Nitride Fuel	--- /No	--- /Pu-U-Nitride Fab.	***Brest, ENHS***	No /No	Yes /Yes	Rx: 6-12 FC: UO_2: >750 Nitride: ~200

Italic – Conceptual design

Note:

1. This is only a crude measure for non-proliferation when it is accounted in this context because not all the inspection activities (such as transport of fuel and spent fuel storage on-site) are included.

The inspection days per year, shown in the last column of Table 5.7 indicate the international effort required to inspect reactor and fuel cycle systems operating with traditional or advanced nuclear fuels. This will be a continuous effort imposed on the owner of the facilities holding nuclear materials regardless whether the facilities are in operation or shutdown. This level of inspection effort could be considered as a quantifiable measure for non-proliferation, as it would be in the interest of the reactor and fuel cycle designers to minimize such burden imposed on their systems.

Waste Management Consideration

Irradiation of advanced nuclear fuel in their respective reactors produces spent fuels with different characteristics. Different burn-ups of these fuels will produce different levels of self-protecting radiation in spent fuel, defined as the time in years a radiation level of 100 r/h

measured in air 1 meter from the surface of the centerline can be maintained. The plutonium content would also vary for different spent fuels upon irradiation. These are shown in Table 5.8.

The spent fuels, or the reprocessed high level wastes (HLW) are required to be disposed of, presumably in a suitable geologic repository. An important consideration for the advanced fuel development is to ensure that the resulting spent fuels or HLW can be accepted for geologic disposal. Comparing to spent UO_2 fuel, the DWPF glass wastes and spent DOE-owned fuel, the acceptance of the spent advanced fuel is intuitionally assessed and listed in Table 5.8. Detailed evaluation of waste acceptance is needed to confirm results of the preliminary assessment.

Table 5.8 Waste Management Measures for Different Nuclear Fuels and Fuel Cycles

Case	Fuel Type	Reactor	Burn-up (GWd/ tonne)	Self-Protecting Radiation in spent fuel (y)	Plutonium content in spent fuel (%)	Waste Acceptance	Remark
Weapon Production	Driver /Production	**Production Reactor**	<10	<5		DWPF glass waste form	Accepted by YM
LWR Once Thru	UO_2	**LWR**	33 - 50	~120	~1	Spent Oxide Fuel	Accepted by YM
LWR w/MOX	UO_2 /MOX	**LWR**	33 - 50	~120	~1 to 4	Spent MOX Fuel	Similar to DOE-owned spent fuel
Candu Once Thru	UO_2	**Candu**	8.3	<5	~0.2 to 0.4	Spent Oxide Fuel	Studied by AECL
Candu /MOX	Canflex	***Candu /MOX***	17	<10	~0.2 to 0.4	Spent Oxide Fuel	Studied by AECL
PBMR, MHTGR	Coated Particle	***PBMR, MHTGR***	80	>120		Carbide Spent Fuel	Similar to DOE-owned spent fuel
IFR, 4S	Metal	***IFR, 4S***	~100	>120	~20	Metallic Spent Fuel	Similar to DOE-owned spent fuel
Brest, ENHS	Nitride Fuel	***Brest, ENHS***	Ave. 150	>120		Nitride Spent Fuel	

Italic – Conceptual design

The aim here is to design fuels such that upon irradiation, the spent fuels can be directly disposed of in a geologic repository with or without chemical processing.

6. FUEL FABRICATION CAPABILITY COST ANALYSIS

LLNL has a glovebox system within Building 332 that has the potential to be converted for use as an advanced fuel fabrication R&D test bed. The plutonium ceramification test facility (PuCTF) was developed by the Immobilization Development and Testing Program for the Fissile Materials Disposition Program (NN-60) as a proof-of-process system to prepare plutonium-ceramic pucks. The Immobilization Program was suspended by DOE in March 2001 and is currently in the process of dismantling and storing or disposing of equipment. The PuCTF glovebox will be dismantled and put into storage pending resumption of the Immobilization effort.

This section presents a scoping study of the modifications, cost, and schedule for converting and activating the PuCTF glovebox for an advanced fuel fabrication application.

Background

The PuCTF was developed as a semi-scale, plutonium-ceramic fabrication glovebox line for prototypical proof-of-process development and testing of the Immobilization plant process. The objectives:

- One half to full-scale equipment prototypes would process plutonium oxide materials reliably and successfully into ceramic pucks, and
- Full-scale plutonium-ceramic pucks could be fabricated successfully for the expected range of impurity materials in plutonium feed through physical measurement of puck properties and through performance (leach) testing of samples from the pucks.

The proposed PuCTF tests were essential to confirming that the conceptual design of the processing equipment for ceramification is correct and to provide essential operational and reliability data for final plant design. It would also provide much needed data for the licensing and qualification of the waste form.

Installation of the PuCTF glovebox system in Room 1345 of the LLNL Plutonium Facility was underway when DOE suspended the Immobilization Program in March 2001. The glovebox and mechanical component installation was essentially complete. Electrical and I&C work was not completed. Thus the PuCTF has not been run hot, i.e., with plutonium. Previously, the PuCTF glovebox system had been completely assembled, checked out, and a series of cold test (without plutonium) runs made. The Immobilization Suspension Plan approved by DOE calls for dismantling and storing the PuCTF glovebox pending resumption of the Immobilization effort. The actual dismantlement and storage is awaiting the availability of plutonium handlers and could begin at any time.

Production of the plutonium-ceramic form has several functional steps in common with the production of mixed oxide (MOX) fuel pellets. If suitable agreements can be developed with NN-60, a modified PuCTF has potential application for R&D of some prototype advanced fuels.

Suitability of PuCTF for advanced fuel fabrication R&D

Table 6.1 compares some of the major process steps for plutonium ceramification and mixed oxide fuel production. The advanced fuel types suitable for investigation in the modified glovebox will have production requirements similar to those for MOX fuel.

Table 6.1 Comparison of functional steps

PuCTF Immobilization Process	Advanced fuel R&D process
Commercial vendor fabricates precursor materials	Commercial vendor provides uranium oxide
Dry mill actinide oxides & co-mill/mix dry with binder in attritor mill (aluminum balls in mill)	Micronization (uranium ball milling)
Granulate milled powder (binder mixing)	Secondary blending, additive mixing
Press granulated feed (hydraulic press)	Press blended feed (hydraulic press)
Sinter	Sinter
-	Dry centerless grinding
NDA/NDE	NDA/NDE

Based on the functional similarities, cold testing of the PuCTF components, and the anticipated performance parameters for making advanced fuel pellets, the following equipment currently in the PuCTF is believed adaptable for advanced fuel development R&D:

- Glovebox
- Attritor (with uranium ball milling in place of the aluminum)
- Granulator
- Press (with die set replaced)
- Material handling equipment up to press

PuCTF equipment which is not expected to have the necessary performance range or operating characteristics and which would be replaced includes:

- Puck (pellet) handling equipment from the press to the furnace and from the furnace to the glovebox bag out
- Sintering furnace
- NDA/NDE equipment

Additional equipment not needed for ceramification and not part of PuCTF but essential to fuel pellet production includes:

- Centerless grinding equipment and glovebox
- Sintering furnace with nitrogen atmosphere

Design, installation, activation, and R&D campaign schedule

A preliminary schedule has been developed to provide the timeline for modifying the PuCTF design, training plutonium handlers as operators, procuring, fabricating, installing, and testing the equipment, obtaining facility, LLNL, and DOE approvals, and conducting operations producing prototype fuel pellets for R&D. The schedule extends over 3 years and assumes:

- The PuCTF is not dismantled and placed in storage
- Plutonium handlers are available for assignment before the end of the first quarter after the project start
- A nominal 3 year window of opportunity is available

Attachment 5-1 presents a preliminary schedule from start of design through several production campaigns, and photos of the ceramification equipment taken within the gloveboxes.

Cost Estimate

An order of magnitude cost estimate was built up from projected labor, equipment, and facility costs and the assumed schedule. Table 5-2 summarizes the costs associated with modification of the existing PuCTF glovebox system to a glovebox suitable for R&D on advanced reactor fuel types. The three-year order of magnitude project cost projection is approximately $11.4M.

Key cost assumptions:

- Year 1 assumed for project start - FY02, PuCTF as-is in Room 1345 of B-332
- Labor escalation at 5.5% in year 2 and 5.8% in year 3
- Equipment costs based on engineering judgement and experience with PuCTF
- Plutonium Facility tax at 35% of project funding level
- Centerless grinding glovebox can be located in Room 1345
- NDE/NDA space will be provided elsewhere in B-332

- Plutonium feedstock and disposal and waste disposal are excluded
- Glovebox D&D are excluded

Table 6.2 Cost summary for modfying the PuTCF to an advanced fuel R&D line ($K)

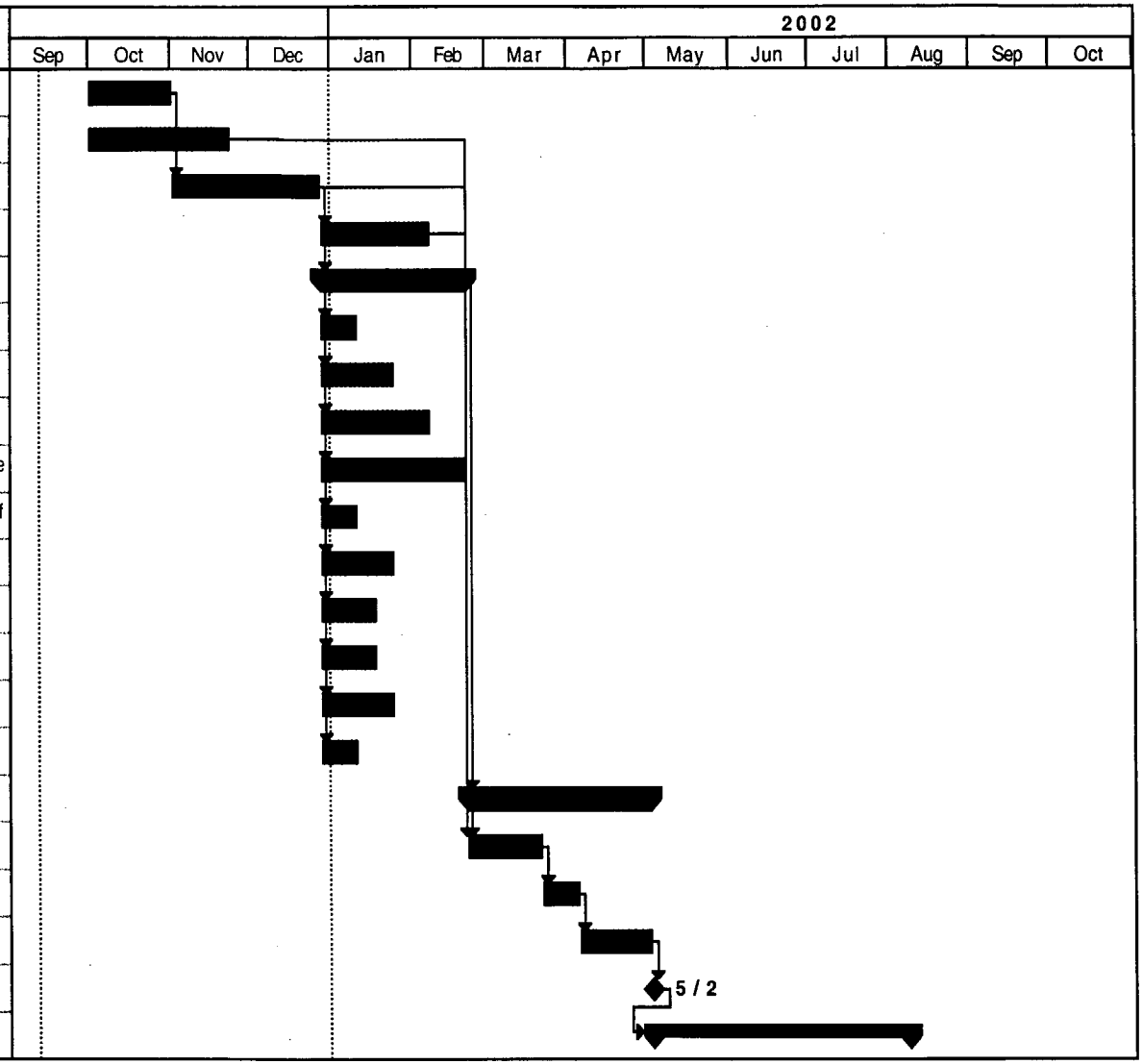
	Year 1	Year 2	Year 3
	Design and Installation	Activation and Operation	Operation and Closeout
Labor	1,916	1,951	1,272
Labor escalation	0	107	74
Materials	995	60	60
Facility operation	1,019	741	492
Progrmmatic (Experimentation, Computations, UC Berkeley, Post Doc., etc.)	470	641	1602
Total	4,400	3,500	3,500

Issues impacting conversion to advanced fuel R&D

- GB is scheduled for removal and storage. If removed, cost increases significantly
- Availability of plutonium handlers is critical path item
- Source of plutonium oxide feedstock assumed available
- Disposition path for acceptable pellets assumed available

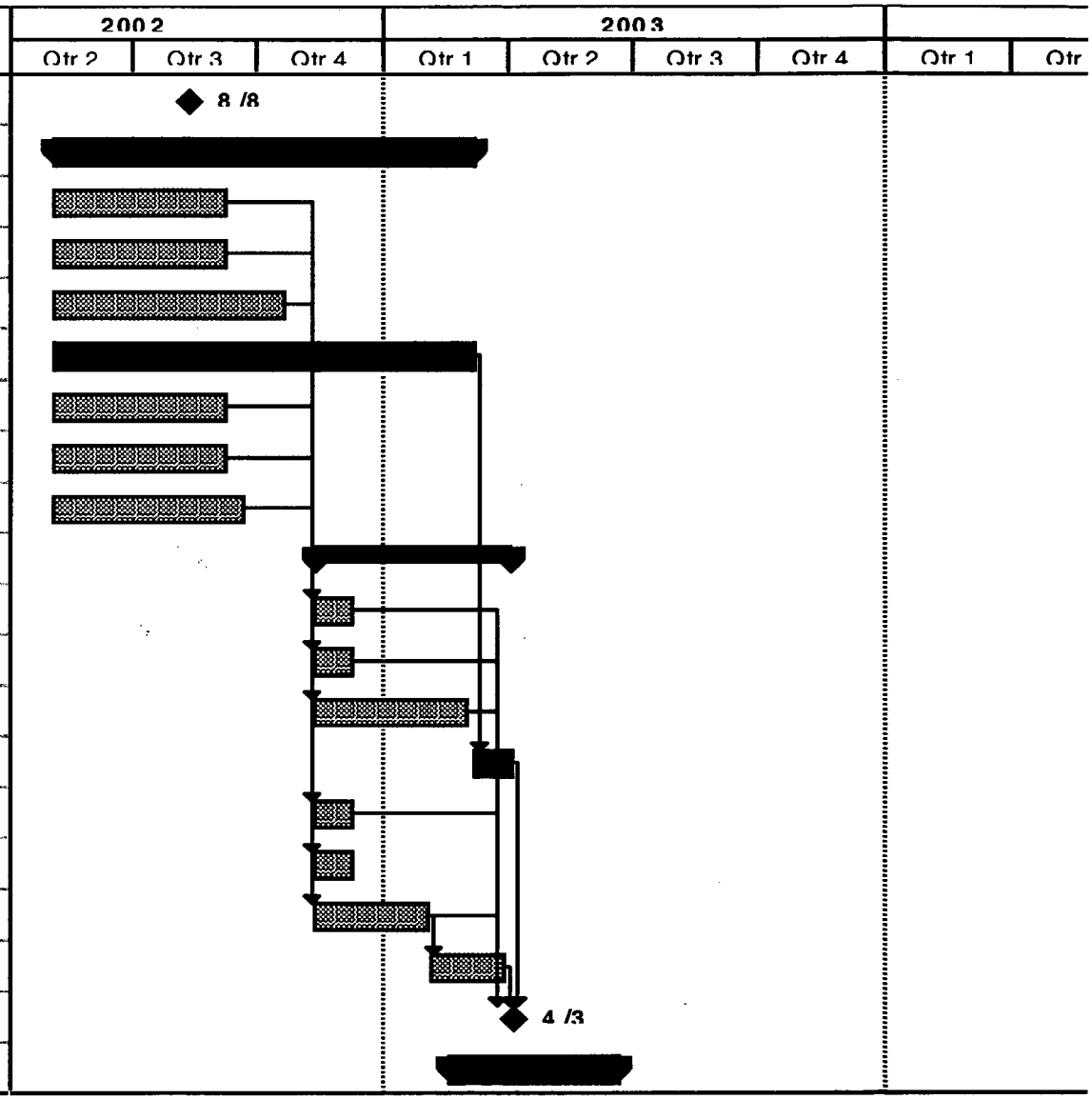
Task definition and initial redesign

ID	Task Name
1	Review Design File
2	Develop preliminary campaign plan
3	Establish Design Basis
4	Downselect attritor, granulator rqumnts
5	**Engineering & Design**
6	Prepare press specification
7	Prepare pellet handling equipment specification
8	Prepare furnace equipment specification
9	Prepare grinding/polishing glovebox & equipme
10	Prepare dust control, recycle equipment specif
11	Prepare NDA/NDE equipment specification
12	Criticality,safety reviews
13	Recycle, waste reviews
14	Revise controls design
15	Revise Glovebox lay out
16	**Complete System Design**
17	Project design review complete
18	FEDR complete
19	Modify designs
20	Design complete
21	**USQ Process**



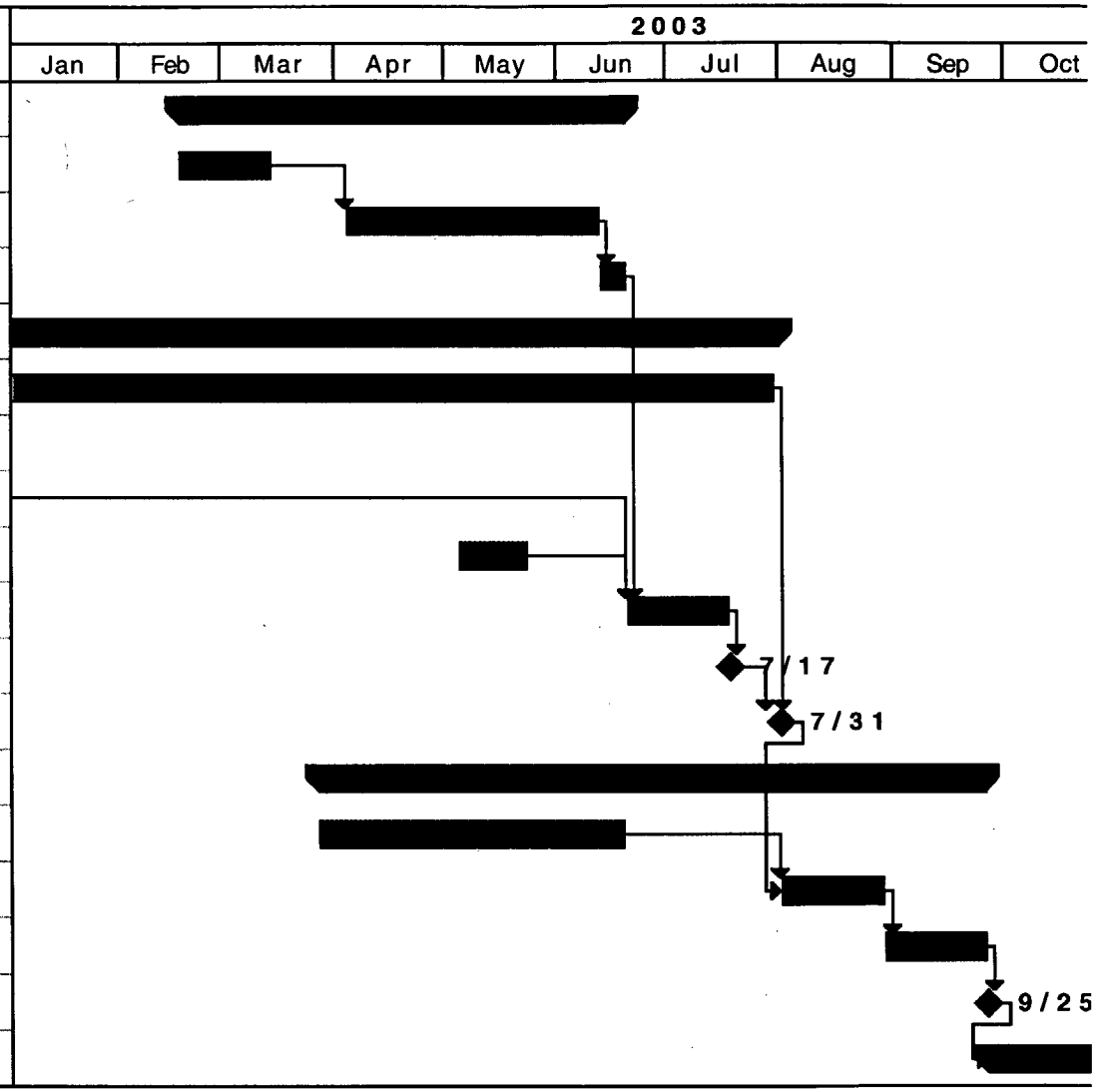
Procure and install equipment

ID	Task Name
24	USQ documentation complete
25	**Procure equipment**
26	press
27	pellet handling
28	furnace & furnace gas
29	grinding/ polishing
30	dust control, recycle
31	NDA/NDE equipment
32	controls
33	**Install equipment**
34	press
35	pellet handling
36	furnace & furnace gas
37	grinding/ polishing
38	dust control, recycle
39	NDA/NDE equipment
40	controls, piping, electrical
41	Complete as-built, P&ID, electrical dwgs
42	Equipment installation complete
43	**Cold Test**



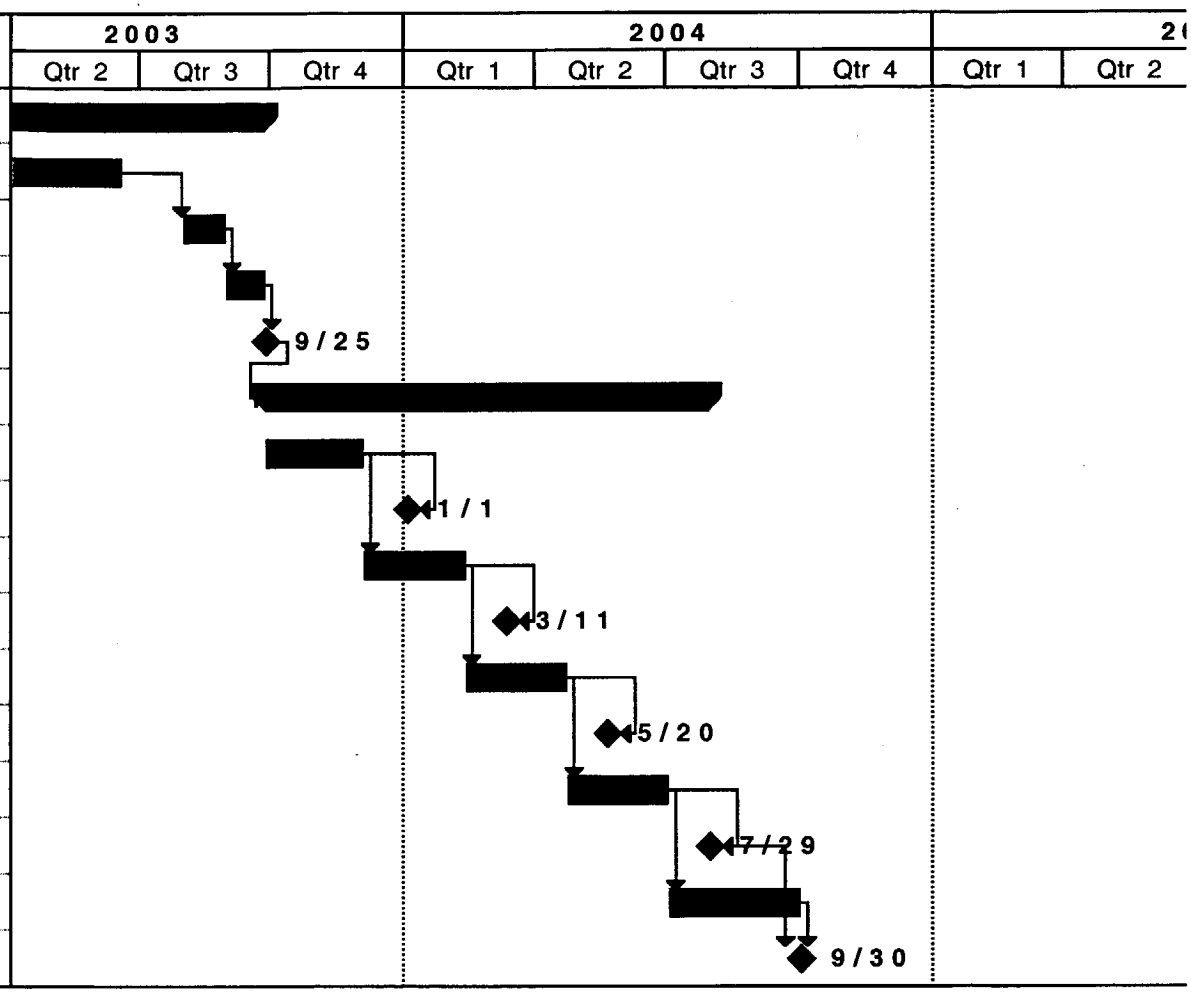
Activate Glovebox

ID	Task Name
43	**Cold Test**
44	Write draft cold test procedure
45	Conduct cold tests and training
46	Modify Procedure
47	**OSP Process**
48	Operator recruit & training
49	PSAP complete
50	Write OSP draft
51	Complete criticality analysis
52	OSP Review
53	Complete OSP
54	Complete 0perator training
55	**ORR**
56	Develop ORR Plan
57	Conduct ORR
58	Review ORR results
59	ORR complete
60	**Hot operations**



Advanced fuel R&D hot operation

ID	Task Name
55	**ORR**
56	Develop ORR Plan
57	Conduct ORR
58	Review ORR results
59	ORR complete
60	**Hot operations**
61	Campaign AFT I
62	Complete Material Characterization Rpt I
63	Campaign AFT II
64	Complete Material Characterization Rpt II
65	Campaign AFT III
66	Complete Material Characterization Rpt III
67	Campaign AFT IV
68	Complete Material Characterization Rpt IV
69	Decon and mothball
70	Complete closeout report



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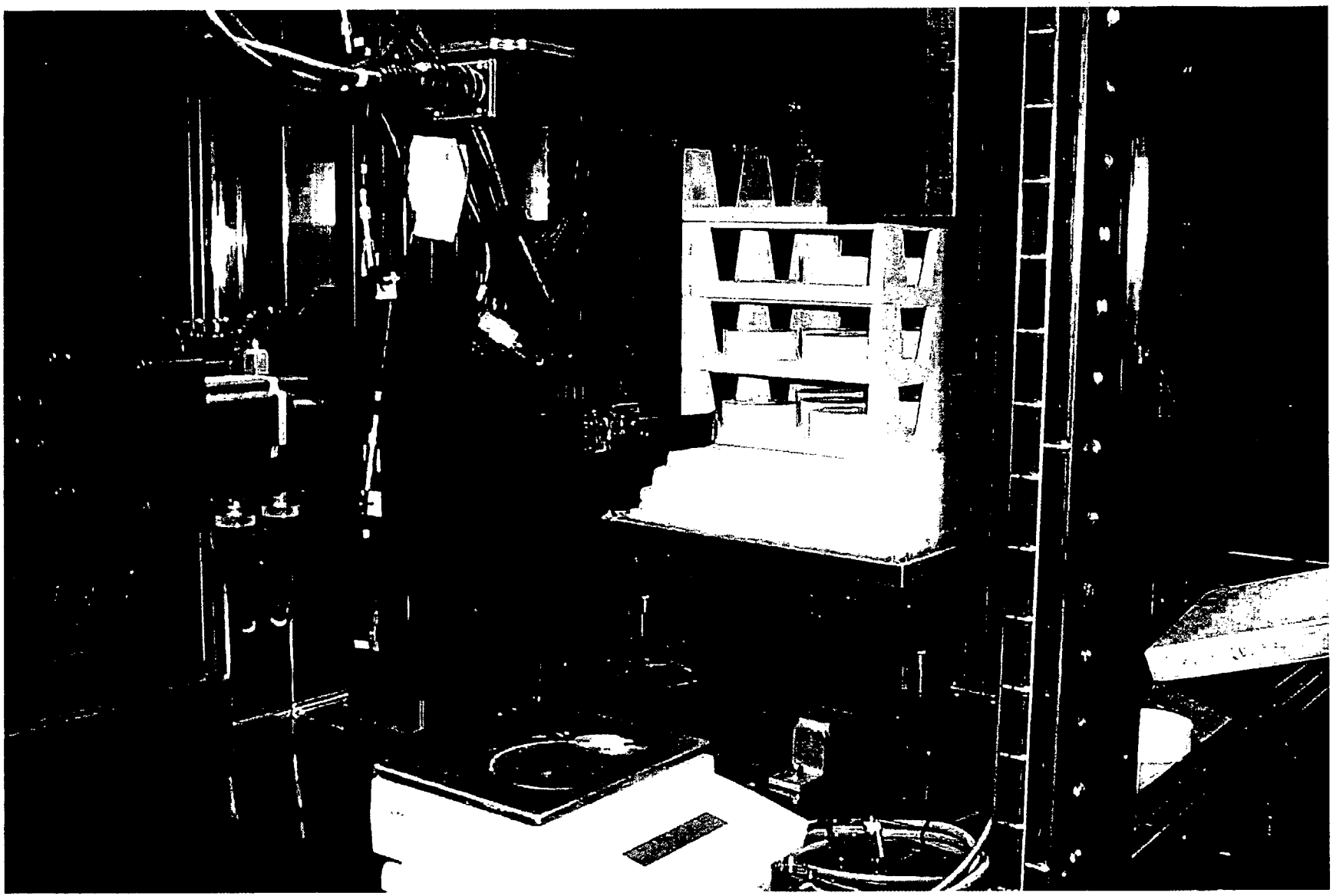

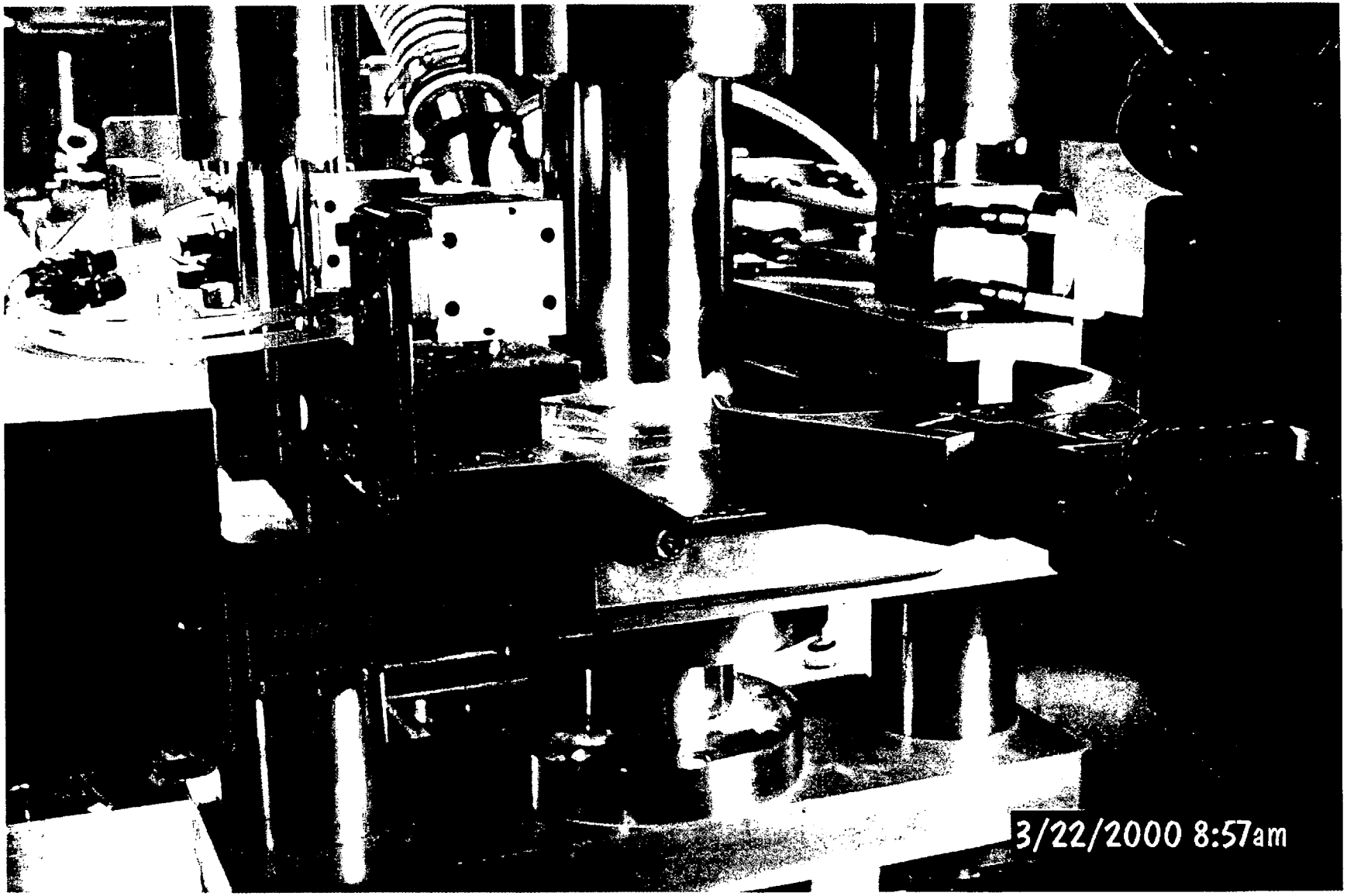
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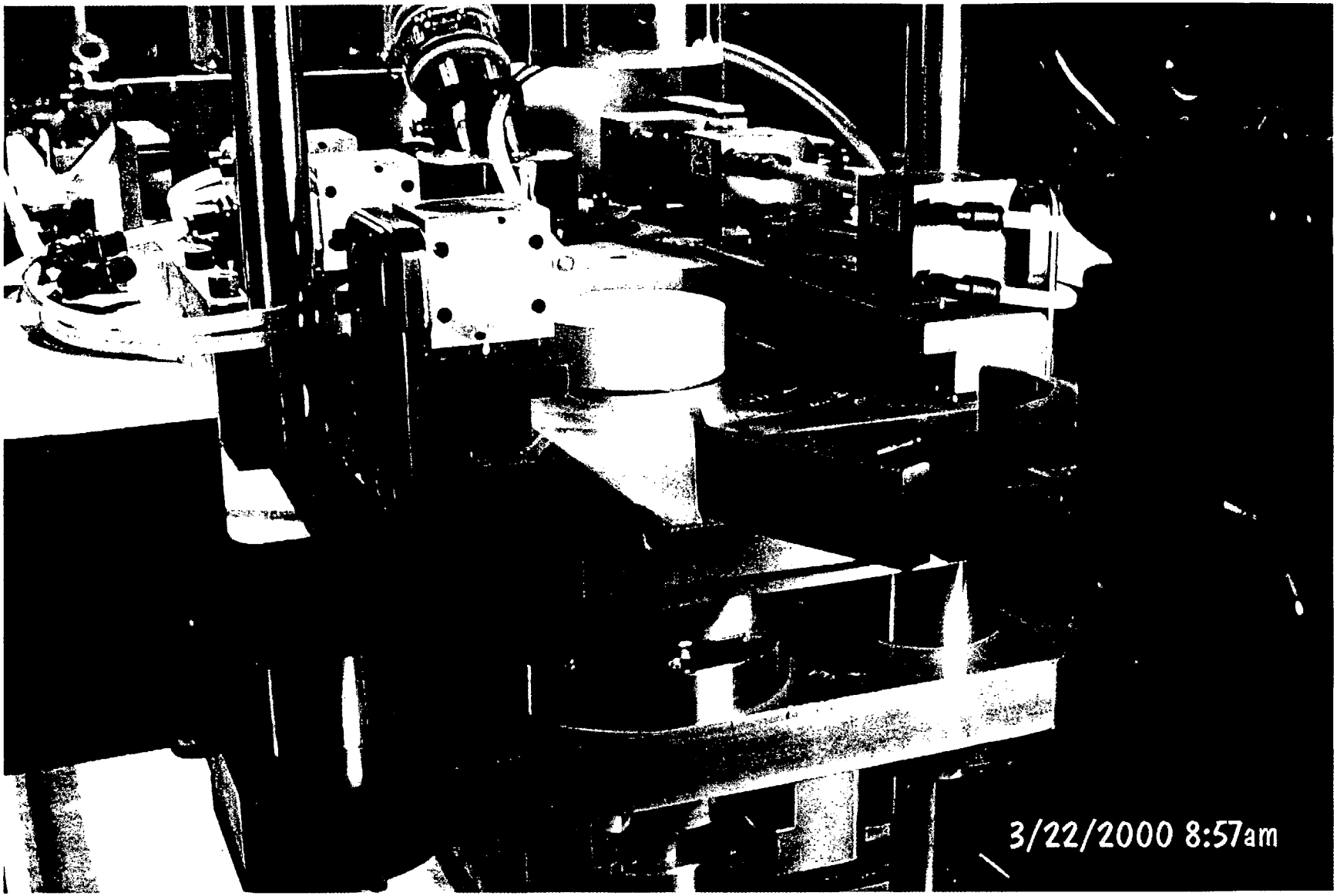
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