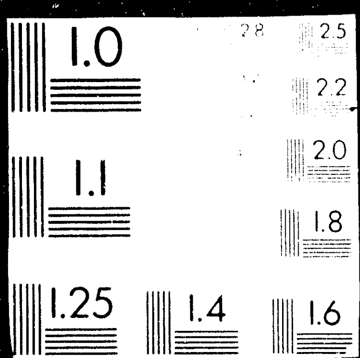
1 OF 1



SAFETY ANALYSIS OF IFR FUEL PROCESSING IN THE ARGONNE NATIONAL LABORATORY FUEL CYCLE FACILITY*

by

I. Charak and D. R. Pedersen
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439

R. J. Forrester and R. D. Phipps
Argonne National Laboratory
P. O. Box 2528
Idaho Falls, Idaho 83403

International Conference and Technology Exhibition on Future Nuclear Systems: Emerging Fuel Cycles and Waste Disposal Options, ANS

Seattle, Washington
September 12-17, 1993

*This work was performed under the auspices of the U. S. Department of Energy, Office of Technology Support Programs, under Contract No. W-31-109-Eng-38.

SAFETY ANALYSIS OF IFR FUEL PROCESSING IN THE ARGONNE NATIONAL LABORATORY FUEL CYCLE FACILITY

I. Charak, R. J. Forrester*, D. R. Pedersen, and R. D. Phipps*
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439
(708) 252-2718/3335

ABSTRACT

The Integral Fast Reactor (IFR) concept developed by Argonne National Laboratory (ANL) includes on-site processing and recycling of discharged core and blanket fuel materials. The process is being demonstrated in the Fuel Cycle Facility (FCF) at ANL's Idaho site. This paper describes the safety analyses that were performed in support of the FCF program; the resulting safety analysis report was the vehicle used to secure authorization to operate the facility and carry out the program, which is now under way. This work also provided some insights into safety-related issues of a commercial IFR fuel processing facility. These are also discussed.

I. INTRODUCTION

The Integral Fast Reactor (IFR) concept developed by Argonne National Laboratory (ANL) for the U. S. Department of Energy (DOE) includes on-site processing and recycling of discharged core and blanket fuel materials. The process is being demonstrated in the Fuel Cycle Facility (FCF) at ANL's Idaho site.

The FCF was designed more than thirty years ago as a key element in the demonstration of an integrated fuel cycle associated with the Experimental Breeder Reactor-II (EBR-II). That fuel cycle comprised the remote processing of the reactor's metal-alloy fuel by a technique known as melt refining, the injection-casting of new fuel pins, the fabrication of stainless-steel clad, sodium-bonded fuel elements using the processed fuel, and the production of fuel subassemblies for insertion into the reactor. The history of the FCF and a complete description of the earlier fuel cycle have been documented by Stevenson.[1]

With the successful demonstration of the original EBR-II fuel cycle in the 1960s, during which more than 400 fuel subassemblies were processed, the missions of the FCF and EBR-II were redirected to support the U. S. fast reactor development program. In the early 1980s, ANL began research that led to an innovative fast reactor concept with an integral fuel cycle similar to the original one embodied in the EBR-II-FCF complex. The new concept became known as the IFR.

*Address for these authors is:
Argonne National Laboratory
P. O. Box 2528
Idaho Falls, Idaho 83403
(208) 533-7332/7269

K:\kowalski\charak\global2.ic

A major element in the development of the IFR fuel cycle, which is based on a plutonium-uranium-zirconium fuel alloy, is the use of electrochemical refining of the spent fuel to remove the bulk of the rare-earth, alkaline-earth, alkali-metal, and noble-metal fission products and to separate the plutonium and other transuranic elements from the uranium. In order to utilize the existing FCF in the IFR fuel-cycle development program, it was necessary to upgrade and refurbish the facility so that it would meet current DOE safety and environmental protection requirements, to the greatest possible extent. The current status of the program is given by Lineberry, et al. in another paper in this conference.[2]

II. OVERVIEW OF FACILITY, PROCESS, AND SAFETY ANALYSES

The FCF consists primarily of two interconnected hot cells, one containing air (the air cell) and the other high-purity argon (the argon cell), in which all operations are conducted remotely. EBR-II fuel subassemblies to be processed are brought into the air cell where they are disassembled into their individual fuel elements. The fuel elements are transferred through an air lock into the argon cell where they are chopped into small segments and loaded into the electrorefiner (ER). The ER products are cathodes either of uranium or plutonium-uranium (mostly plutonium) that are relatively fission-product-free. Subsequent steps in the process include consolidation of the ER cathode products (to purify the heavy metal of process chemicals), injection casting of fuel pins, cutting the fuel pins to size, loading the fuel pins and bond sodium into new cladding tubes, and welding an end-plug closure onto the cladding. The new fuel elements are then transferred back to the air cell where they undergo non-destructive examination and are assembled into new subassemblies for reinsertion into the reactor.

The FCF safety analyses addressed potential accidents that were organized along the following lines:

- events resulting from facility equipment malfunctions
- breaches in argon-cell confinement boundary
- events resulting from air-cell equipment malfunctions
- events occurring in support facilities
- events associated with transfer of materials between the air and argon cells.

For the most part, the abnormal events studied resulted in the release of radioactive material to the environment; in one case, the release of cadmium, a process chemical, was the result.

III. DOSE GUIDELINES

In order to assess the significance of radiological accidents, it was necessary to develop guidelines against which to measure accident consequences. The starting point for this development was the likelihood-dependent dose guidelines for the general public prepared by the Los Alamos National Laboratory (LANL) under DOE sponsorship.[3] The LANL guidelines are shown in Table 1. The indicated accident likelihoods are listed in order of decreasing probability, with the last defining the boundary between credible (i.e., design-basis) and incredible (i.e., beyond-design-basis) events. The range of guideline values shown for the last two categories suggests that the dose guideline is a function of the numerical probability of the accident. For FCF, accident probabilities were determined subjectively, so that the public guideline values for the unlikely and extremely unlikely classifications were chosen as the higher of the two end points for each classification. This choice was based on the fact that those doses correspond to limits established by authoritative sources. In particular, for the unlikely classification, the 5 mSv value corresponds both to the DOE permissible (with authorization) temporary annual public limit for routine releases[4] and to the DOE limiting value of annual dose equivalent to an unborn child for an occupational worker.[5] For the extremely unlikely case, the 250 mSv value is the well-known whole-body dose limit embodied in 10 CFR 100.[6]

TABLE 1. LANL Accident Consequence Guidelines

Likelihood Classification	Guideline,[a] mSv
Anticipated	< 0.1
Unlikely	0.1 to 5
Extremely Unlikely	5 to 250

[a]Guideline for whole-body dose from Ref. 3.

Since for FCF we also estimated accident doses to employees, corresponding dose guideline values for occupational workers were necessary. They were developed using the following rationale.

For anticipated events, the worker guideline dose was based on the reasonable principle that it should be no higher than the DOE annual limit of exposure for workers from routine releases,[5] i.e., 50 mSv. Implicit in the current DOE dose limits is the expectation that workers might receive higher radiation doses from routine operations than the public, even though both groups are considered to be adequately protected. It seemed reasonable, therefore, that in an accident situation as well, workers might be expected to receive higher doses than the public. This logic required that a dose limit for workers for extremely unlikely events analogous to the 250 mSv public limit, but higher, be established. Such a limit was derived from the International Commission on Radiological Protection (ICRP) statement that workers who receive whole-body doses of no greater than 1000 mSv "are almost certain to recover."[7] Consequently, at the extremely-unlikely end of the accident-classification spectrum, the worker threshold dose was taken as 1000 mSv. This value is four times the corresponding value for the public; the intermediate value was derived using the same ratio of worker-to-public dose from routine releases as implied in the DOE orders, i.e., 50.

Notwithstanding the fact that the guideline values discussed above were based on reasonable applications of existing authoritative data, the final guidelines were conservatively established by multiplying each of those values, except for the public anticipated-classification value, by 0.2. The reason why we did not apply the same adjustment factor to the 0.1 mSv value is that that value is already ten times smaller than the DOE limit for the public due to routine releases;[4] any added conservatism was unjustified. The resulting FCF guidelines are given in Table 2.

TABLE 2. Fuel Cycle Facility Radiological Dose Guidelines

Classification	Committed Effective Dose Equivalent Guideline, mSv	
	Public	Worker
Anticipated	0.1	10
Unlikely	1	50
Extremely Unlikely	50	200

IV. ACCIDENTS ANALYZED

A wide spectrum of abnormal events (or accidents) was analyzed to show that the operation of FCF presented no undue risk to the public or the operating staff. These analyses consisted of developing a scenario for each accident considered, estimating radiological doses at the site boundary and at the on-site bus-staging area, and comparing those doses with the above guidelines. In addition, the collective subjective engineering judgments of project staff were tapped to categorize each accident by likelihood of occurrence. The following paragraphs summarize some of the accidents analyzed in the final safety analysis report (FSAR).

A. Fission Gas Release from Argon Cell

There are several events that might lead to the abnormal release of fission gas, primarily ^{85}Kr, from the argon cell without actually breaching the cell boundary. These are:

1. Over-pressurization of the argon cell due to loss of cell cooling and subsequent heatup of cell atmosphere, with the pressure buildup relieved by activation of the safety exhaust system (SES).

2. Over-pressurization of the argon cell due to inadvertent activation of argon gas supplies to the cell and subsequent release of cell gas to the environment through the normal cell pressure control system or the SES.

3. Loss of normal electrical power which has the same qualitative effect as item (1) above, except there is a time delay for activating the SES using the emergency power supply.

4. Accidental evacuation of cell gas using a vacuum pump intended for gas-lock purging.

Each of the above events was analyzed; however, the dose results are bounded by the doses calculated for the assumed release of all the ^{85}Kr in the cell at the time of the equipment malfunction(s). That quantity was calculated using conservative estimates of (a) the facility throughout (50% greater than planned), (b) fuel burnup (80% greater than planned), and (c) normal cell atmosphere purge rate (1/3 smaller than expected), and amounts to about 7×10^{13} Bq.

B. Breach in Argon Cell Boundary

The processing of hot heavy metals in the argon cell leads to the possibility of spontaneous ignition and a metal fire if sufficient air leaks into the cell. A number of potential accidents could result in a cell boundary reach. However, a design-basis earthquake (DBE) was identified as a potential initiator that might result in the largest breaches in the argon cell, thereby bounding the consequences of accidents involving smaller breaches. The SES is a safety-grade system that satisfies the DOE requirement to filter airborne particulate products that result from this postulated event. In addition, as a defense-in-depth (DID) measure, confinements for individual process furnaces are designed to survive a DBE, so that only pyrophoric heavy metal outside DID confinements is presumed to be at risk of burning following a DBE.

C. Waste Can Drop

Process wastes from the electrofining furnace and from other process streams are handled in the FCF argon cell in sealed cans, known as waste cans. In handling the waste cans, the possibility of a can being dropped and breached and its contents dispersed cannot be ruled out. The path of a loaded and sealed waste can is normally from the argon cell to the air cell, and eventually into a container through a bagged-transfer device that maintains an atmosphere seal between the air cell and the transfer cask in the tunnel beneath. During the handling of the waste can within the air cell, some handling mishaps can be assumed to occur in which the can loses its integrity, resulting in the exposure of its contents and the release of some fraction of plutonium and fission products from a dropped salt-waste can or of cadmium from a dropped metal-waste can. The contents of the ER salt-waste can were conservatively estimated using Waste Isolation Pilot Plant (WIPP)[8] criteria and considering several possible operating scenarios. The cadmium content of a metal-waste can was maximized at a value consistent with material transfer equipment capacity.

D. Waste Evaporator Rupture

The radioactive liquid waste processing system includes a pot evaporator located in a decontamination chamber. A scenario can be conceived in which multiple filters in the evaporator exhaust line become plugged and pressure-relief is blocked while the liquid feed to the unit and evaporation continue. With the assumed failure of all controls and safety devices and lack of human intervention, the pressure inside the evaporator increases until it ruptures, dispersing the contained radioactivity throughout the chamber. Such an event was analyzed for the FSAR.

Since the plan is to ship the loaded evaporator to WIPP for disposal, it was assumed that the evaporator and its contained radioactive materials satisfy the current WIPP criteria.[8] Those criteria require, inter alia, that the dose rate at the package surface be no greater than 10 Sv/h. However, operational requirements dictated that the surface dose rate should not exceed 3 Sv/h; this was conservatively increased to 5 Sv/h, resulting in a substantial safety margin relative to the WIPP criteria.

E. Subassembly Rupture after Dropping Cask

The interbuilding cask (IBC) is used to transport intact subassemblies between EBR-II and the FCF. In the course of handling the IBC en route to the FCF air cell entry port, it is possible though extremely unlikely that some equipment or personnel failure could cause the IBC to drop, possibly damaging the contained subassembly. The drop could be as much as 4.3 m, causing fuel cladding failures due either to mechanical shock or loss of coolable geometry. Such an event has never happened at the FCF in more than 7,000 such transfer operations.

The source term used in the analysis of this accident was based on a spent fuel subassembly that was irradiated to 18% burnup and cooled for about 100 days to reach a power level that is expected to permit cooling by natural processes when the subassembly is transferred into the air cell for disassembly. Such a cooling time also meets the facility requirement for a minimum of 90 days of cooling, which is intended to reduce the iodine and xenon accident source terms to negligible amounts. Because of the substantial structure of the IBC, some credit was taken for local confinement of the fission products and actinides before leaking into the building.

F. Other Accidents

The accidents described above are examples of the kinds of analyses that were necessary in order to assure that the FCF could be operated safely. A number of other accidents were also considered but are not reported here in detail. They are:

- loss of subassembly forced cooling in air cell
- dropped subassembly in air cell
- subassembly stuck in air cell transfer port
- spent fuel elements or waste can stuck in small transfer lock
- waste box fire
- air cell exhaust system flow reversal

- fuel element failure during washdown.

Except for the accident involving a heat-generating source becoming stuck in the small transfer lock, all of the above abnormal events resulted in the release of radioactive materials to the environment. The analyses required the development of source terms for each case and the calculation of downwind doses using standard techniques for atmospheric dispersion estimates.[9]

G. Accidental Criticality

Worldwide, only eight criticality events have ever occurred in process systems, i.e., outside reactors and critical experiments.[10] Each accident involved aqueous process streams. The IFR processing scheme employs a series of non-aqueous batch operations in which fissionable materials are easily accounted for at each step.

The primary FCF criterion for criticality safety is

> Process designs shall incorporate sufficient margins of safety such that even if a credible combination of two unlikely, independent, and concurrent changes in process conditions occur, the process is still subcritical.

Those familiar with the so-called double-contingency principle of criticality safety[11] will recognize that the FCF criterion is much more restrictive. The FCF project has shown by analysis that credible two-error scenarios meet the above criterion. In general, the errors considered in the analysis involve:

- Mass: Errors resulting in mass overload.
- Material: Substitution of wrong material.
- Density: Average fissionable material density exceeding the design value.
- Reflection: Unanticipated increase in neutron reflection.
- Geometry: Arrangement into a more reactive geometry.

These possible errors were divided into two categories, unlikely and extremely unlikely, on the basis of the following factors: process operation, equipment design, criticality barriers, administrative controls, and operational history.

For each zone or piece of equipment, the analyses showed that during the course of any planned operation, the system remains subcritical for any credible combination of two unlikely abnormal conditions, or the combination of an unlikely with an extremely unlikely, but still credible, abnormal condition occurring without operator intervention.

Because of the above considerations, we believe that a criticality accident in the FCF and in a successor commercial processing facility can be made "incredible", i.e., beyond the design basis.

V. ACCIDENT CONSEQUENCES

Doses at the site boundary (5 km distant) were calculated either for a two-hour exposure following a time-dependent release or an "infinite" exposure following a "puff" release of radioactive material to the environment, as applicable. They are summarized in Table 3 and are clearly a very small fraction of the above guidelines.

TABLE 3. Summary of Fuel Cycle Facility Design Basis Accidents and Doses at Site Boundary

Accident	Classification[a]	Committed Effective Dose Equivalent[b]
^{85}Kr release due to equipment malfunctions[c]	A U E	2×10^{-4}[d]/2×10^{-1} 8×10^{-5}[d]/8×10^{-3} 1×10^{-5}[d]/2×10^{-5}
Breach in argon cell boundary	E	7.6×10^{-3}/1.5×10^{-2}
Loss of subassembly forced cooling	U	7.8×10^{-5}[e]/7.8×10^{-3}
Dropped subassembly	A	3.8×10^{-5}[d]/3.8×10^{-2}
Subassembly stuck in transfer port	E	7.8×10^{-5}[e]/1.6×10^{-4}
Waste can drop[f]	E	2.9×10^{-4}/5.8×10^{-4}
Spent fuel or waste can stuck in small transfer lock	NA	0[g]
Waste box fire	U	1×10^{-6}/1.0×10^{-4}
Waste evaporator rupture	E	8×10^{-4}/1.6×10^{-3}
Subassembly rupture within cask	E	1.6×10^{-2}[e]/3.2×10^{-2}
Air cell exhaust system flow reversal	E	1.3×10^{-2}/2.6×10^{-2}
Fuel element failure during washdown	A	3.8×10^{-5}[d]/3.8×10^{-2}

[a]A = anticipated; U = unlikely; E = extremely unlikely; NA = not applicable - no release
[b]mSv/percent of FCF guideline
[c]Values are highest for each classification
[d]Effective submersion dose
[e]Does not include submersion dose
[f]Cadmium release from a metal-waste can results in trivial off-site consequences
[g]No release expected

VI. SAFETY CONSIDERATION FOR A COMMERCIAL IFR FUEL PROCESSING PLANT-LESSONS LEARNED FROM FCF

The safety issues (and/or risk) associated with the operation of a commercial IFR fuel processing facility cannot be quantified at this time, since no such facility has yet been designed. However, experience derived from the safety analysis of ANL's FCF, the subject of this paper, allows one to predict with confidence that such issues will be inconsequential, both in an absolute sense and by comparison with its associated reactor(s). Some of the factors that lead to this conclusion are discussed below.

A. Source Term

A commercial facility will contain only a small fraction of the fission products and heavy metal contained in a liquid metal reactor (LMR). With adequate (though not excessive) cooling time prior to processing, the iodine and xenon source terms can be virtually eliminated, thereby eliminating the major accident contributor to thyroid exposure and a significant contributor to external doses, respectively. In addition, the solid fission product accumulation in the electrorefiner will likely be limited for process reasons, so that the solid fission product inventory potentially at risk will be likewise limited. Moreover, if the commercial process develops along lines that are similar to the one being demonstrated in FCF, the quantity of solid fission products potentially at risk will be even more limited, since the electrorefiner provides a level of confinement such that the fission products are not considered available for release in an accident.

B. Energy Source

The fuel processing facility does not require high-energy sources, so that there is no driving force to disperse radioactive materials to the environment. Moreover, as noted above, the process is batch-oriented, so that there is no need for process lines to penetrate the containment (or confinement). All process stages can be contained within a massive hot cell, which will be designed to withstand all appropriate natural phenomena, such as earthquakes and tornadoes.

C. Batch Size

The batch size used in the FCF are intended to be prototypic of a commercial facility. For example, the FCF injection casting furnace batch produces 25 kg of fuel alloy; a batch smaller than this size, assuming two batches per day (or one batch per day in each of two furnaces), was shown in an earlier assessment to be capable of supporting a liquid metal reactor (LMR) capacity of 1,400 MWe. This relatively small batch size is reflected throughout the facility and demonstrates that the amount of heavy metal potentially at risk is a small fraction of the loading in a reactor.

D. Containment

Although no specific design has been proposed, the commercial IFR fuel cycle facility readily lends itself to a "double-containment" concept for accident mitigation. The first structural barrier to the release of radioactive materials would be the safety-grade process cell itself, which will be a low-leakage structure designed to accommodate the forces usually associated with a reactor secondary containment. The cell would be entirely within a process building, which will also be a safety-grade structure with a ventilation system that provides enormous dilution of cell leakage, followed by filtered exhaust to the environment. A third level of confinement could be provided by the process equipment itself.

E. Criticality Safety

The small batch sizes noted above support the contention that a criticality accident can be made "incredible" by a combination of equipment design and administrative controls. The FCF analyses described earlier showed that the likelihood of an accidental criticality in the process equipment due to gross operating errors and equipment malfunctions was beyond the design basis. For the commercial facility, many of the consequences of similar malfunctions would likely be even more benign, since the FCF fuel alloy consists of plutonium and highly enriched uranium for makeup, whereas the commercial IFR fuel would use natural or depleted uranium as makeup.

F. Accident Mechanisms

For the FCF safety analysis, a broad spectrum of accidents was analyzed. These are, of course, strongly dependent on the details of the facility and process design. However, it was concluded that there was no accident that could release sufficient quantities of fission products and actinides to cause any concern at the DOE-controlled site boundary (i.e., exclusion area boundary). In fact, the worst accident caused a dose at the site boundary more than two orders of magnitude smaller than the guideline value (cf. Table 3). Since the FCF is more than thirty years old and has to be backfitted to accommodate the fuel cycle development program, it was not feasible to take advantage of the kinds of features that would likely be included in a new licensed commercial facility, such as the double containment identified above.

From the above qualitative assessment is it judged that the risk of operating a commercial IFR fuel cycle facility can be made so low as to be inconsequential compared with the reactor(s) it services, or indeed compared with a PUREX-type reprocessing facility. Not included in this assessment is an analysis of the risk associated with the handling of high-level radioactive waste out of the facility for ultimate disposal. Such an assessment is dependent not only on the process design but on the criteria imposed by the commercial disposal facility.

ACKNOWLEDGMENTS

This work was performed under the auspices of the U. S. Department of Energy, Office of Technology Support Programs, under Contract No. W-31-109-Eng-38.

REFERENCES

1. C. E. STEVENSON, The EBR-II Fuel Cycle Story, American Nuclear Society (1987).

2. M. J. LINEBERRY et al., "Status of IFR Fuel Cycle Demonstration," these proceedings.

3. J. C. ELDER et al., "A Guide to Radiological Accident Considerations for Siting and Design of DOE Nonreactor Facilities," LA-10294-MS, Los Alamos National Laboratory (January 1986).

4. U. S. DEPARTMENT OF ENERGY, "Radiation Protection of the Public and the Environment," DOE Order 5400.5 (February 8, 1990).

5. U. S. DEPARTMENT OF ENERGY, "Radiation Protection for Occupational Workers," DOE Order 5480.11 (December 21, 1988).

6. U. S. NUCLEAR REGULATORY COMMISSION, "Reactor Site Criteria," Title 10, Code of Federal Regulations, Part 100, Section 100.11, as amended (April 1962).

7. INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION (ICRP), "The Principles and General Procedures for Handling Emergency and Accidental Exposures of Workers," ICRP Publication 28, Pergamon Press, New York (1978).

8. WESTINGHOUSE ELECTRIC CORP., "TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant," WIPP-DOE-069, Revision 3 (January 1989).

9. U. S. NUCLEAR REGULATORY COMMISSION, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, Revision 1 (November 1982).

10. W. R. STRATTON, "A Review of Criticality Accidents," Revised by D. R. Smith, DOE/NCT-04 (March 1989).

11. AMERICAN NATIONAL STANDARDS INSTITUTE, INC., "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983 (1983).

END

DATE FILMED

11/16/93