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Public Health Risks of Substituting Mixed-Oxide For Uranium Fuel in Pressurized-Water Reactors

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The U.S. Department of Energy (DOE) has awarded a contract to the consortium Duke Cogema Stone and Webster (DCS) to dispose of up to 33 tonnes of excess weapons-grade plutonium (WG-Pu) by irradiating it in the form of mixed-oxide (MOX) fuel in four U.S. commercial pressurized-water reactors (PWRs). This paper estimates the increase in risk to the public from using WG-MOX at these reactors and finds that it exceeds recently established Nuclear Regulatory Commission (NRC) guidelines. Therefore, the NRC will have a technical basis for prohibiting the use of MOX at these reactors unless the risk that they will experience a severe accident can be significantly reduced.

MOX fuel will displace a fraction of the low-enriched uranium (LEU) fuel that these reactors currently use. Because MOX cores have greater quantities of plutonium and other actinides than LEU cores throughout the operating cycle, the source term for radiological releases caused by severe reactor accidents will be greater for MOX-fueled PWRs. In this paper, the radiological consequences to the public from containment failure or bypass accidents at MOX-fueled PWRs are calculated, and compared to those resulting from the same accidents at LEU-fueled PWRs.

This paper finds that compared to LEU cores, the number of latent cancer fatalities (LCFs) resulting from an accident with core melt and early containment failure would be higher by 39%, 81% or 131% for full WG-MOX cores, depending on the fraction of actinides released (0.3%, 1.5% or 6%). Under the DCS plan, in which WG-Pu will be purified using an aqueous process and only 40% of the core will be loaded with WG-MOX, the number of LCFs would be 11%, 25% or 30% higher, respectively. The average LCF risk to individuals within ten miles of a severe accident approximately doubles for a full WG-MOX core, and increases by 26% for a DCS core.

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These results are of particular concern for the nuclear plants in the DCS consortium, Catawba and McGuire. These plants have ice-condenser containments, which Sandia National Laboratories estimates are at least two orders of magnitude more vulnerable to early failure than other types of PWR containments.

The findings of this paper also apply to the proposed use of WG-MOX in VVER-1000 reactors in Russia, which meet less stringent safety standards than U.S. reactors.

INTRODUCTION

Plutonium Disposition

In January 1997, the U.S. Department of Energy (DOE) decided to pursue a "dual track" policy for disposing of approximately 50 tonnes of plutonium produced for weapons programs that have been declared excess to military needs. The two tracks refer to different approaches for converting separated plutonium into a dilute and highly radioactive form that is more difficult to return to weapons.

Under one approach, known as "can-in-canister" immobilization (CIC), plutonium will be incorporated into chemically stable ceramic discs. These discs will in turn be embedded in canisters of "vitrified" (glassified) high-level radioactive waste (VHLW) at the Defense Waste Processing Facility (DWPF) at the Savannah River Site (SRS) in South Carolina. DOE is planning to use CIC for approximately 17 tonnes of excess plutonium in impure forms. The CIC facility will be sited at SRS adjacent to the DWPF.

Under the other approach, plutonium will be used to produce "mixed plutonium-uranium oxide" (MOX) fuel assemblies, which will be irradiated in a number of U.S. commercial light-water nuclear reactors (LWRs), displacing some or all of the low-enriched uranium oxide (LEU) fuel the reactors currently use. DOE is planning to utilize this option for 25.6 tonnes of weapons-grade plutonium (WG-Pu).

Both processes are regarded by most experts as roughly comparable in their ability to render the plutonium as inaccessible as the plutonium in commercial spent nuclear fuel, thereby meeting the "spent fuel standard" defined by the National Academy of Sciences (NAS).[1] However, DOE decided to pursue both tracks for a number of reasons, one being the desirability of having a backup strategy in case one approach did not succeed.

In 1998, DOE issued a Request for Proposals, seeking vendors interested in providing MOX fuel fabrication and irradiation services. Of the three proposals submitted, two were quickly eliminated for failing to meet basic requirements. In March 1999, DOE signed a contract with the third party, a consortium called Duke Cogema Stone & Webster (DCS), which included the

U.S. utilities Duke Power and Virginia Power, the French national fuel cycle company Cogema, and the Stone & Webster architect-engineering firm.[2]

According to the contract, DCS will design, build and operate a MOX fuel fabrication plant at SRS. MOX fuel will then be irradiated in six pressurized-water reactors (PWRs) at three sites --- Virginia Power's North Anna plant, located about 70 miles from Washington, DC, and Duke Power's McGuire and Catawba plants, both situated within twenty miles of downtown Charlotte, North Carolina. In January 2000, DOE confirmed this plan in its Record of Decision (ROD) on the Surplus Plutonium Disposition Final Environmental Impact Statement. However, in April 2000, Virginia Power withdrew from the consortium, leaving only four reactors in the MOX program.

Use of MOX fuel on a large scale will be a novel practice for a U.S. nuclear utility. Although some countries in Europe have begun using MOX fuel on a limited basis in LWRs, U.S. utilities have not followed suit. This is a result both of the U.S. non-proliferation policy adopted in the late 1970s (and reaffirmed by the Clinton Administration) which led to moratoria on commercial spent fuel reprocessing and MOX recycling, and by the poor economics of MOX fuel, which is several times more expensive than LEU.

Environmental Impacts of MOX Use

The immobilization and MOX approaches, which both require large-scale handling and processing of plutonium, will be expensive and will pose risks to human health and the environment. However, these risks are likely to be small in comparison to those that were encountered when the material was produced.

Many arms-control advocates believe that the costs of plutonium disposition are justified by the security benefits. Some observers have argued further that differences in cost and risk between the two disposition tracks are not important considerations,[3] a view that does not take into account fiscal and political constraints. Cold War-sized budgets are not being made available for disarmament activities, and the plutonium disposition program is under pressure to minimize costs. Also, many environmental groups and citizens' groups near affected sites oppose disposition activities unless they have low environmental and public health impacts.

Cost and public health impact were indeed major considerations in the process DOE used to select MOX and immobilization from the large number of disposition options that were initially proposed. In deciding on the dual track policy, DOE argued that there were no significant differences between the

MOX and immobilization options with regard to these criteria. However, its analysis of this issue was inadequate.

To accurately compare the environmental impacts of the two disposition approaches, one must determine the increase in risk associated with modifying existing processes to accommodate plutonium disposition. The MOX approach consists of several stages, each of which can have a significant environmental and public health impact. A plant for fabrication of MOX fuel will be built and operated, and the fuel will be shipped to reactor sites and irradiated. Afterward, the spent MOX fuel will have to be stored on site until a geologic repository becomes available.

By comparison, the environmental impacts of CIC immobilization result primarily from operation of the ceramic immobilization plant. This plant will be similar to the MOX fabrication plant in many ways, and it will have similar (if not lower) impacts. Moreover, the CIC process is being designed and tested to ensure that there is no impact on the safe operation of the DWPF, and preliminary results have been encouraging.[4] The risks associated with MOX transportation to reactor sites and irradiation are not encountered in the CIC process. Thus, a comparative analysis of the two options will not be complete without an assessment of the risks of MOX irradiation.

MOX Use and Severe Accident Risk

Probabilistic Risk Assessment

The tools of probabilistic risk assessment (PRA) can be used to estimate the risk to the public from the operation of nuclear power plants with WG-MOX fuel. PRA consists of three steps:[5]

- 1. Identification and delineation of the combinations of events that, if they occur, could lead to an accident (or other undesired event);
- 2. Estimation of the chance of occurrence for each combination; and
- 3. Estimation of the consequences associated with each combination.

Total risk is then defined as the product of the probability and the consequences of each combination, summed over all combinations.

PRAs are carried out in three stages. A Level 1 PRA identifies all sequences of events that could result in core damage and estimates their frequencies of occurrence. Summing the frequencies of these events yields the core damage frequency (CDF), the average annual probability that core dam-

age will occur. A Level 2 PRA evaluates containment performance for each accident sequence that leads to core damage, and calculates the radiological release for each sequence in which containment is compromised. A Level 3 PRA estimates the consequences (i.e. prompt fatalities and latent cancer fatalities among the public) of the sequences involving radiological releases, and combines all elements into a measure of the total risk to the public. Level 3 PRA analysis uses computer codes that model radionuclide dispersal and radiation exposure of the population. These calculations require as input radionuclide release fractions obtained from Level 2 PRA, which are highly uncertain in some cases.

PRAs are huge, complex calculations and their results contain large uncertainties. Therefore, they are more useful for ranking the relative risk significance of different events than for providing meaningful values of absolute risk. Consequently, the U.S. Nuclear Regulatory Commission (NRC) has been slow to incorporate PRA-based analysis into its regulations, although it has recently adopted a policy to increase use of PRA (see below).

At the request of the NRC, all U.S. nuclear plants conducted Level 1 and 2 PRAs for "internal events" for the Individual Plant Examination (IPE) program.[6] Subsequently, "external" events (seismic, tornado and fire risks) were analyzed for the Individual Plant Examination of External Events (IPEEE). While the NRC collected and analyzed the IPE results, it did not peer-review or endorse them.[7] In fact, because the methodology and assumptions varied widely from plant to plant, the overall results were inconsistent in important respects. In the future, PRAs used in licensing will have to conform to quality-control and peer-review standards now under development.

Most plants have not completed credible Level 3 PRAs. The NRC itself conducted peer-reviewed Level 3 PRAs for five U.S. nuclear plants and presented the results in the 1990 report NUREG-1150.[8]

It will not be a simple undertaking to conduct PRAs for MOX-fueled LWRs. In general, the substitution of WG-MOX for LEU fuel will affect both the probability of occurrence and the consequences of each reactor accident sequence, requiring extensive modification of LEU-based PRAs. The difficulty will be compounded by the relative lack of experience with the use of WG-MOX fuel and the absence of data on many technical aspects of MOX use.

The task can be simplified by focusing on the reactor accidents that are the largest contributors to the overall risk to the public: the "beyond-design-basis" or "severe" accidents that involve extensive core damage and failure (or bypass) of the reactor containment. An important subset of severe accidents are those that result in large radiological releases before the local population has evacuated, increasing the chances for early health effects. If the use of

MOX fuel were to result in an increase in the consequences and/or probabilities of these accidents, then the total risk to the public would increase by a similar percentage.

The most difficult part of this assessment is the calculation of the probabilities of severe accidents at MOX-fueled plants. To do this accurately, the computer codes used by U.S. utilities and the NRC to analyze accidents will have to be modified to incorporate WG-MOX-specific parameters.[9] Some of these parameters have not been fully benchmarked even for RG-MOX fuel, for which there is considerably more experimental data. Nonetheless, the NRC has stated that the use of MOX "will not have a big effect on accident progression" and consequently that "it appears likely that the probability of severe accidents will not change."[10] Consistent with this assessment, this paper assumes that severe accident probabilities are the same for LEU and MOX cores.

In contrast, assessing the consequences of severe accidents at MOX-fueled plants is straightforward, especially under the assumption that MOX and LEU fuel rods behave similarly under severe accident conditions.[11] In that case, the radionuclide release fractions (the fractions of the reactor core radionuclide inventories that are released during an accident) are the same for both types of fuel, and the differences in consequences are due entirely to their different radionuclide inventories.

By assuming that severe accident probabilities and radionuclide release fractions are the same for both LEU and MOX cores, risk calculations are simplified. Level 2 PRA results for LEU cores can be adjusted for MOX cores by using the appropriate radionuclide inventories. This is the approach used here.

However, there are technical aspects of the use of MOX fuel that can affect the validity of these two assumptions. This issue is discussed in more detail below.

Impact of MOX Fuel on Accident Consequences

Throughout the operating cycle, MOX cores have larger inventories than LEU cores of most transuranic (TRU) radionuclides, including plutonium-239 (Pu-239), americium-241 (Am-241) and curium-242 (Cm-242). Since many of these radionuclides are long-lived alpha-emitters, with relatively high radiotoxicities if inhaled or ingested, small releases during an accident can contribute significantly to public radiation exposure.

Inventories of fission products are also different in LEU and MOX cores, because U-235 and Pu-239 have slightly different fission product spectra. Also, at least one TRU isotope, Np-239 (a beta-emitter), has a lower inventory

in a MOX core. However, the results presented below indicate that these differences are less significant for risk than the increase in alpha-emitters.[12]

DOE's February 1996 Storage and Disposition of Weapons-Usable Fissile Materials Draft Environmental Impact Statement (DPEIS) did not analyze the environmental impacts of accidents at MOX-fueled LWRs, but analyzed an LEU-fueled LWR instead. DOE justified this by claiming that

"...studies ... indicate that the use of MOX fuel in a ... LWR does not increase the risk and consequences of accidents. This results from the fact that the other radioisotopes that are released in an accident have more serious impacts on human health than the Pu used in the MOX fuel."[13]

This is based on the assumption that the consequences of reactor accidents are dominated by releases of the more volatile fission products, such as iodine-131 (I-131) and cesium-137 (Cs-137), while plutonium and other actinides, which typically have very low vapor pressures (are "low-volatile"), would not be released to the environment in significant quantities.

However, DOE's statement is not consistent with the current state of understanding of severe accidents. Although most actinides are low-volatile and are not easily released from molten fuel, significant actinide releases into the environment can occur during certain accidents. Although these accidents are expected to occur very infrequently, there are both historical precedents and regulatory requirements for considering them in safety analyses.

The Chernobyl accident has demonstrated that significant and wide-ranging dispersal of low-volatile radionuclides is possible. A recent review of the Chernobyl source term has concluded that the release fraction for actinides was approximately 3.5%. Moreover, dispersal of these relatively heavy aerosols was not limited to the immediate vicinity of the plant; fuel fragments were discovered as far away as Greece and Germany, over one thousand kilometers away.[14]

One often hears the claim that a Chernobyl-type accident cannot happen in the West because Western reactors have robust containment structures, and the particular accident sequence that occurred was specific to Chernobyl-type (RBMK) reactors. However, while the presence of a containment dome at Western reactors reduces the risk of such accidents, it does not eliminate it entirely. Analysts have identified hypothetical accident sequences at U.S. LWRs which can lead to energetic mechanical dispersal of the fuel, catastrophic failure or bypass of the containment and significant releases of low-volatile core fragments in the form of aerosols.

Uncertainties in the low-volatile release fractions predicted to result from severe core damage range over several orders of magnitude. The NRC estimates that low-volatile releases as high as several percent of the core inven-

tory can occur, and uses this information in its risk studies.[15]

One mechanism for core dispersal and early containment failure at PWRs is known as high-pressure melt ejection (HMPE), in which the reactor vessel fails at high pressure after a core melt. The molten fuel is then dispersed into the containment in a shower of fragments, resulting in direct containment heating (DCH), a very high rate of heat transfer to the containment atmosphere. The resulting rapid pressurization of the containment can cause it to fail. A phenomenon of even greater concern is the buildup of hydrogen from the reaction of zirconium cladding and water, which can also lead to an explosion capable of fragmenting fuel and breaching the containment.

The Nuclear Control Institute (NCI) cited these issues in its comments on the DPEIS, and DOE subsequently revised its analysis. The December 1996 Final Programmatic Environmental Impact Statement (FPEIS) estimated that using a full core of MOX in an existing LWR would change the number of latent cancer fatalities (LCFs) resulting from a severe accident by +8% to -7%: in other words, the number of LCFs could actually decrease as a result of switching to MOX fuel. However, DOE's calculation contained a number of flaws which cast doubt on its accuracy.[16] For this reason, NCI undertook the present study.[17]

This paper finds, contrary to DOE's assertions, that there are significant public health risks associated with the MOX approach. This is due primarily to the finding that the consequences of severe accidents at MOX-fueled LWRs will be greater than those at LEU-fueled LWRs, as a result of the larger inventories of plutonium and other actinides in MOX cores.

The risk of containment failure and energetic fuel dispersal is of particular concern for the reactors that have been selected for the U.S. MOX program. All four reactors have ice-condenser containments, which are considerably smaller and weaker than the large dry containments present at most U.S. PWRs. A recent study, conducted for the NRC by Sandia National Laboratories (SNL), concluded that “ice condenser plants are at least two orders of magnitude more vulnerable to early containment failure than other types of PWRs.”[18] The study cites the McGuire plant in particular as having an unacceptably high probability of containment failure. Commenting on this finding, Dana Powers, Chairman of the NRC Advisory Committee on Reactor Safeguards (ACRS), said that the selection of McGuire for MOX irradiation was “maybe a suboptimal choice.”[19]

Impact of MOX Fuel on Accident Probabilities

The risk analysis in this paper assumes that the use of MOX fuel will not affect the probability that a severe accident will occur. While this assumption is reasonable as a first approximation, there are a number of differences in the neutronic and thermomechanical properties of the two types of fuel that could affect the outcome of accident precursors. Not all of the negative impacts of MOX fuel use can be mitigated by modifications to the core design. The question of whether the remaining differences will have a significant effect on the likelihood of a severe accident remains open and will likely require considerable experimental and analytical work to resolve.

Pu-239 has higher thermal absorption and fission cross-sections than U-235, resulting in a less thermal neutron spectrum in MOX cores. Compared to an LEU core, a WG-MOX core will have[20]

- a more negative moderator temperature coefficient
- a generally more negative Doppler coefficient
- reduced delayed neutron fraction and prompt neutron lifetime
- reduced individual control rod worth
- reduced boron worth
- increased local power peaking factors
- higher centerline fuel temperatures

To permit the use of full-core MOX, while preserving margins of safety such as the shutdown margin,[21] modifications must be made to the reactor core, such as an increase in the number and/or quality of rod cluster control assemblies (RCCAs). For cores with a MOX loading no greater than 33%, additional control rods may not be absolutely necessary if one is willing to accept a large reduction in the shutdown margin. Although DCS has found that the shutdown margin is minimal for a 40% MOX loading, it has stated that it does not intend to install additional control rods in McGuire and Catawba, although it may change the type of control rod material used in McGuire. In contrast, PWRs in France that use 30% MOX cores have four additional RCCAs installed.

MOX cores also require a greater concentration of the neutron-absorbing isotope B-10 in the coolant, which can be accomplished by increasing the boric acid concentration or by using boron that has been enriched in B-10. Because

there is a limit on the amount of boric acid that can be added to the coolant without causing problems during operation, DCS has said that it is considering the use of enriched boron.[22]

Power peaking, caused by the large differences in flux between adjacent MOX and LEU assemblies, also must be taken into account in partial MOX core design. This requires use of zoned fuel assemblies containing fuel pins with several different plutonium enrichment levels, as well as careful placement of MOX assemblies in the core. Even with these modifications, power peaking still can occur due to the heterogeneous distribution of plutonium within the MOX fuel pellets. (Variations in isotopic content can also result in significant power peaking, although this is more of a concern with RG-Pu.)

Another property of MOX fuel that could affect radionuclide release during accidents is the "inferior physical behavior of MOX pellets to that of UO_2 pellets" due to "higher temperature and greater fission gas release," especially at burnups greater than 35 GWD/t.[23] The heterogeneous microstructure of MOX fuel (resulting from the presence of plutonium clusters) results in the appearance of "hot spots" of very high local burnup, in which high concentrations of fission gas accumulate.[24] During reactivity insertion accidents (RIAs), this gas can be released suddenly, causing fuel rod cladding failure and fuel pellet fragmentation. This problem is exacerbated by the fact that MOX fuel has a lower thermal conductivity and a higher centerline temperature than LEU fuel.

Based on the results of a series of RIA tests at the Cabri reactor in France, French regulators have concluded that "MOX fuel shows a higher failure potential than UO_2 at comparable burnup"[25] and that there is a "very high potential for rupture" of MOX fuel during RIAs.[26] In one experiment, a MOX fuel rod segment with a burnup of 55 GWD/t (which is typical of burnups attained in U.S. PWRs today) experienced a violent rupture and dispersal of fuel particles at a peak enthalpy of 120 cal/g, while an LEU rod of comparable burnup was able to withstand a similar peak enthalpy without rupture. Although some high-burnup, highly corroded LEU rods also failed during the test series, researchers concluded that the failure of the MOX rod was unique because it did not have a heavily corroded cladding.

Unresolved safety questions of MOX fuel performance have led French regulatory authorities to restrict the irradiation time of MOX fuel assemblies to three annual cycles, or an assembly-averaged burnup of 41 GWD/t, whereas the LEU limit is 52 GWD/t. As a result, French plants with partial MOX cores continue to operate on annual cycles while other plants have been able to switch to 18-month cycles, which lowers their costs by reducing the frequency of refueling outages. Even at lower burnups, however, the Cabri test series

found that fission gas release in MOX fuel was significantly higher than in LEU fuel.

More evidence of problems with MOX fuel has surfaced at another fuel test facility in France known as VERCORS, where it was observed that during the early stages of core degradation, "releases of volatile radionuclides from MOX are more extensive than from conventional fuels at similar levels of burnup" which is "consistent with the peculiar nature of porosity that develops in MOX during burnup."[27] In particular, in a test in which spent fuel was held at a temperature of 1780 K for one hour, the cesium release fraction for a MOX fuel rod with a burnup of 41 GWD/t was 58%, compared to only 18% for an LEU rod with a burnup of 47 GWD/t.[28]

The Cabri tests suggest that the probability that an RIA will cause significant fuel damage and progress to a severe accident may be greater when MOX fuel is in the core. Therefore, the probabilities of some severe accidents may increase when MOX is substituted for LEU in PWRs, unless strict burnup limits are imposed on MOX assemblies.

The VERCORS experiments suggest that MOX radionuclide release fractions may need be changed to take into account the greater observed releases of volatile radionuclides. This would result in more severe consequences of MOX-fueled accidents than are estimated in this paper.

DCS has stated that initially it plans to irradiate MOX fuel for only two 18-month cycles, with limits on assembly-averaged and peak rod burnups of 45 GWD/t and 50 GWD/t, respectively.[29] However, peak *pellet* burnups in this case will exceed 55 GWD/t, the region of concern identified in the Cabri test. In fact, the MOX rod that failed during the Cabri test was taken from a fuel assembly with an average burnup of 46 GWD/t. Moreover, DCS has said that it intends to eventually irradiate MOX for three 18-month cycles, which would require raising the MOX peak rod burnup to over 60 GWD/t, well outside of the current experimental database.

Nuclear Regulatory Issues

In order to obtain regulatory approval for using MOX fuel, DCS will have to apply to the NRC for amendments to its reactor operating licenses. A detailed understanding of the increased risks associated with MOX fuel will be important for the license amendment process.

This paper proposes an approach for judging the safety of MOX use in LWRs according to the "risk-informed" regulatory procedures now being adopted by the NRC.[30] In July 1998, the NRC issued Regulatory Guide (RG) 1.174, which is one of the first systematic applications of PRA in its regulations.[31] RG 1.174 provides a methodology for ranking proposed changes to a

nuclear plant according to their risk significance. Changes which are determined (by PRA methods) to substantially increase risk to the public will be subject to greater regulatory attention by the NRC. Notably, RG 1.174 defines an upper limit for acceptable levels of risk increase.

In January 2000 the Commission granted NRC staff the authority (in special cases) to use risk-informed analysis, such as RG 1.174 guidelines, in its reviews of license amendment requests.[32] If the NRC determines that a proposed license amendment would result in an unacceptable increase in the risk of a severe accident, it will be able to reject the request or attach significant conditions to its approval, even if the request satisfies all regulatory requirements.

RG 1.174 can be used to evaluate the regulatory implications of the increased risk associated with use of MOX fuel. The calculations presented in this paper indicate that the DCS MOX plan is likely to cause an increase in risk to the public that will exceed the upper limit specified in RG 1.174. This should provide a basis for the NRC, in the course of MOX license amendment reviews, to require a comprehensive analysis of the severe accident risk to the public from the use of MOX fuel. To fulfill this requirement, DCS will have to conduct MOX-specific Level 3 PRAs for Catawba and McGuire, subject to rigorous standards of quality control and peer review. If the magnitude of the risk increase estimated below is confirmed by this analysis, the NRC will have justification to require significant modifications to, or even reject, the DCS MOX license amendment requests.

Calculation of MOX Severe Accident Consequences

Core Inventories

This paper adopts a standard approach, using the SAS2H/ORIGEN-S computer code, to calculate LEU and WG-MOX core radionuclide inventories at the end (EOC) and beginning (BOC) of an equilibrium cycle. SAS2H/ORIGEN-S is a module of the Oak Ridge National Laboratory (ORNL) SCALE 4.3 code which generates burnup-dependent cross sections and simulates fuel irradiation.[33]

Three cases were analyzed. The first is based on DOE's Surplus Plutonium Disposition Draft EIS (DEIS), which assumed use of a dry process for converting plutonium pits to oxide and full MOX reactor core loadings.[34] The full MOX core design was adapted from a 1996 Westinghouse report written under contract for DOE, which NCI acquired under the Freedom of Informa-

tion Act (FOIA).[35] DOE commissioned Westinghouse to design MOX cores that were compatible with current PWR operating cycles. Accordingly, the core management strategy assumes a typical 18-month cycle with three-batch reloading, using Westinghouse VANTAGE-5 fuel.[36]

The second case is based on the core design proposed by DCS in March 1999 after it was awarded the MOX contract.[37] The DCS plan uses reactor cores loaded with 40% MOX, as well as solvent extraction ("polishing") to remove gallium (a metal used to stabilize the crystalline structure of plutonium in nuclear weapons) and other impurities from the WG-Pu oxide received from DOE.[38] (DOE initially had assumed that aqueous purification would not be necessary.)

Finally, although not directly applicable to the WG-Pu disposition program, the inventory of a typical reactor-grade (RG) MOX core was calculated. Given the greater quantities of actinides like Am-241 and Cm-242 in irradiated RG-MOX fuel, as well as the higher plutonium loadings necessary to achieve adequate reactivity, one would expect the consequences of severe accidents at PWRs using RG-MOX to be even more severe than at those using WG-MOX.

Table 1 lists the core design parameters used for the three cases. The specifications were based on the Final Safety Analysis Report (FSAR) of a typical four-loop Westinghouse PWR,[39] with appropriate modifications for the MOX core drawn from the Westinghouse report cited previously.[40] Although the Westinghouse report does not provide designs for partial MOX cores at equilibrium, the DCS core inventories could be derived from full-core results. The cycle lengths were adjusted in both cases to achieve an average discharge burnup of 44 GWD/t.[41] Most of the DEIS parameters were unchanged in the DCS case.[42]

Table 1: LEU and WG-MOX Core Characteristics

Design Parameter	**LEU**	**DEIS/DCS WG-MOX**
Number of feed assemblies	88	92
Total number of assemblies	193	193
Feed Loading (tHM)	37.2	38.9
Core total loading (tHM)	81.6	81.7
Average feed enrichment (w/o)	4.25 (U-235)	4.54/4.43 (WG-Pu+Am-241)
Total Pu used per year (t)	0	1.16/0.46
Average discharge burnup (MWD/THM)	44,080	44,080
Cycle length (MWD/THM)	20,100	21,010
Cycle length (effective full-power days)	460	481
Outage length (days)	40	40
Core power (MW_{th})	3565	3565
Soluble boron B-10 content (a/o)	19.8	40
Mid-cycle boron concentration (ppm)	1100	550

Table 2 specifies the isotopic compositions of the fresh fuel assemblies. The WG-Pu isotopic compositions were derived from the reference composition in the DOE DEIS.[43] In the DCS case, the calculation assumes that three years will elapse between the Pu polishing step and the loading of the MOX fuel into the reactor. Immediately after polishing, the concentration of Am-241 will be reduced from about 0.8 weight-percent to a few parts per million. However, it then will steadily increase as a result of Pu-241 decay.

Table 2: Initial Isotopic Composition of Fuel

	LEU	WG-MOX (DEIS/DCS)	RG-MOX
Uranium composition (w/o)			
U-234	0.04	0.002	0.002
U-235	4.25	0.2	0.2
U-236	0.01	0.001	0.001
U-238	95.7	99.797	99.797
Plutonium composition (w/o)			
Pu-238	--	0.04/0.04	2.3
Pu-239	--	92.37/93.08	56.2
Pu-240	--	6.49/6.54	24.2
Pu-241	--	0.24/0.21	9.0
Pu-242	--	0.1/0.1	6.9
Am-241	--	0.76/0.03	1.4

Table 2 also displays the composition of the RG-Pu used in the RG-MOX calculation, which assumes that the RG-Pu is obtained from LEU spent fuel irradiated to a burnup of 44 GWD/t and cooled for nine years before reprocessing, and the RG-MOX fuel is stored for three years before being loaded in the reactor, the assumptions used by the French utility Electricité de France (EDF).

Other details needed for the calculation are provided in Tables A.1 and A.2 of Appendix A.

Results

The inventories of Pu and other actinides in MOX cores are functions of the isotopic content of the Pu in the fresh fuel, the initial Pu core loading and the irradiation time. At EOC, the DEIS core contains, compared to the LEU core, quantities of Pu-239, Am-241 and Cm-242 that are greater by factors of 3, 7 and 7, respectively. At BOC, these factors are 11, 22 and 13.

In the DCS core at EOC, with a smaller MOX core fraction and lower initial Am-241 content, the inventories of most transuranics will be about 2 times greater than those in the LEU core.[44]

Results of the calculation for fifty-eight radionuclides are presented in Tables A.3-A.5 of Appendix A.

Consequence Assessment

Using the core radionuclide inventories tabulated in Appendix A, one can calculate the radiological consequences of severe accidents. The NRC-approved MACCS2 code[45] was used to estimate the consequences of three severe accidents for LEU, DEIS and DCS cores. A representative four-loop PWR with an ice-condenser containment was chosen for analysis.

The indicators used to measure accident consequences are (1) the total number of expected latent cancer fatalities (LCFs) among the population within a thousand miles of the plant, (2) the total number of prompt fatalities (PFs) caused by acute radiation exposure, and (3) the average LCF risk to an individual within ten miles of the plant.[46] The calculation only considered radiation exposures received during a one-week emergency phase following the accident.

The MACCS2 database was updated by replacing the outdated dose conversion factors in the model by the most recent compilation of the International Commission on Radiological Protection (ICRP), ICRP Publication 72.[47] These factors are based on a revised model of the human respiratory tract which reduces the dose from inhalation of actinides.[48] However, other internal features of the MACCS2 code may result in underestimates of the radiological risk associated with exposure to alpha-emitters.[49]

In the calculations, generic parameters were used for population and atmospheric data.[50] Since the quantities of most interest are the ratios of results for WG-MOX and LEU cores, more detailed site characterization was not necessary. While the absolute values of the accident consequences depend strongly on these parameters, the consequence ratios do not. Varying these parameters had only a small effect (less than 10%) on consequence ratios.

Severe accident source terms

The MACCS2 code requires as input a *source term*, the specific radionuclide release that occurs during an accident. The source term is usually expressed as a set of *release fractions* --- the fraction of the core inventory of each radionuclide that is released to the environment. The timing and magnitude of the release of a particular radionuclide from failed fuel rods is determined by the plant conditions associated with each accident sequence and the thermochemical properties of the radionuclide.

Volatile and semi-volatile radionuclides, such as the noble gases, iodine, tellurium and cesium, are usually released from melted fuel in the form of gases or very fine aerosols that can easily escape into the environment through a breach in the containment. In contrast, most actinide metals and oxides have extremely high melting and boiling points, and will be

released in large quantities only if a violent explosion causes mechanical dispersal of the molten uranium oxide core. Nevertheless, mechanical dispersal of fuel is possible during severe accidents.

To simplify the analysis of accident consequences, radionuclides with similar thermochemical properties are grouped together in a *release class*. All members of a release class are assumed to behave identically during the course of an accident. Pu is assigned to the cerium (Ce) group and Np, Am and Cm are assigned to the lanthanum (La) group.[51]

Table 3 displays a set of simplified severe accident source terms which were derived from the NRC's PRA for the Sequoyah plant.[52] Sequoyah, a four-loop Westinghouse PWR with an ice condenser containment, is very similar to the DCS Catawba and McGuire plants. The Sequoyah analysis was used because peer-reviewed PRAs for Catawba and McGuire are not publicly available.[53]

Table 3: Source Terms For Consequence Calculations

Source term	Time of Release (h)	Release duration (s)	Radionuclide release fractions							
			Kr	I	Cs	Te	Sr	Ru	La	Ce
ST-1	5.5	200	1	0.37	0.27	0.13	0.025	8E-3	1.6E-3	8E-3
(f_1=2.8E-7)	6.0	7200	0	0.22	0.35	0.30	0.13	3E-3	0.013	0.018
ST-2	6.0	200	1	0.05	0.04	0.02	4E-3	1E-3	2E-4	1E-3
(f_2=3.6-E6)	6.06	7200	0	0.13	0.15	0.11	0.045	1E-3	5E-3	6E-3
ST-3	1.0	1800	1	0.075	0.06	0.02	5E-3	1E-3	3E-4	1E-3
(f_3=3.1E-6)	1.5	7200	0	0.04	0.06	0.05	0.02	6E-4	3E-3	3E-3

These source terms are associated with the three different plant (core and containment) damage states that contribute to the risk of large early releases. Each one corresponds to a severe accident that results in loss of core cooling, core melt, breach of the pressure vessel and radiological release to the environment. Each source term comprises two plumes, the first consisting of a short duration pulse corresponding to breach (or bypass) of the containment and the second, longer pulse corresponding to subsequent releases from core-concrete interactions.

The three events are distinguished by the timing and mode of radionuclide release to the environment. ST-1 and ST-2 are representative of “very early” and “early” containment failures, occurring prior to and concurrent with breach of the pressure vessel, respectively. ST-3 is a containment bypass

event, in which release to the environment occurs through breach of a steam generator or another barrier between the primary coolant system and systems outside of containment. The probability of very early containment failure (ST-1) is associated with hydrogen combustion, and for PWRs it is thought to be significant only for ice-condenser containments.

The sum of the frequencies of all the accidents corresponding to each source term in Table 3 equals the frequency of occurrence *f*, which is the expected probability per year of an accident resulting in the specified radionuclide release. (These values take into account "internal initiating events" only --- that is, they do not consider the possibility of earthquakes, floods, high winds or fires.)

The three source terms in Table 3 account for all severe accidents that are associated with a significant risk of PFs and LCFs (from early exposures). The sum of their frequencies equals the "large early release frequency" (LERF), which is defined in NRC RG 1.174 as "the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects."

All other accidents fall into two other plant damage states: either "late" containment failure (twenty-four hours after the start of the accident), or no containment failure. Neither of these has the potential for causing early fatalities; they therefore do not contribute to the LERF. Moreover, their contribution to the overall risk to the public of latent cancer fatalities is negligible compared to early containment failure or bypass accidents. The only exception is for late containment failure, in the unlikely event that a significant fraction of the nearby population had not been evacuated a full day after the start of the accident.

Results

Results of the MACCS 2 calculations show that higher transuranic inventories in MOX cores lead to increased severe accident consequences.[54] Table 4 presents the results obtained by averaging the LCFs caused by severe accidents at EOC and BOC. The cycle-averaged number of LCFs is 81% to 96% greater for a DEIS core (corresponding to 1,440 to 6,165 additional LCFs), and 20% to 25% greater for a DCS core (corresponding to 350 to 1,855 additional LCFs). The EOC and BOC results can be found in Tables B.1 and B.2 of Appendix B.

Table 4: Consequences of severe reactor accidents (cycle average).

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-1					
Latent cancer fatalities	7,565	13,730	9,420	1.81	1.25
Prompt fatalities	62	88	64	1.42	1.03
ST-2					
Latent cancer fatalities	2,515	4,920	3,080	1.96	1.22
Prompt fatalities	20	28	21	1.40	1.05
ST-3					
Latent cancer fatalities	1,735	3,175	2,085	1.83	1.20
Prompt fatalities	33	28	29	0.85	0.88

The fractional increase in LCFs is approximately four times smaller for the DCS core than for the DEIS core, reflecting both the reduced MOX core loading and the effect of Am-241 removal from Pu polishing.[55] For a fixed core loading, removal of Am-241 reduces the number of LCFs by about 10%.

Smaller increases (around 40% for the DEIS core and 5% for the DCS core) occur in the number of PFs resulting from early containment failures. For containment bypass, the number of PFs was found to decrease by 12% (from 33 to 29). However, the total number of PFs is approximately a hundred times smaller than the number of LCFs.

Sensitivity analysis

A sensitivity analysis was carried out to assess the impact of varying the actinide release fractions.

The three PWR source terms presented in Table 3 are based on the mean values of distributions of experimental and analytical data for the release fractions in each class. Use of mean values is appropriate for volatile and semi-volatile radionuclides, because the range of uncertainty is believed to be within an order of magnitude. For low-volatile radionuclides like the actinides, however, the release fractions can vary by several orders of magnitude, depending on details of the accident progression. For this reason, the MACCS2 calculation was repeated for a range of actinide release fractions.

Three source terms were compared (Table 5). ST-M is the same as ST-1 of Table 3, with actinide release fractions on the order of 1%. ST-H and ST-L, which incorporate high and low values of the actinide release fractions, were

based on the limits of the data distributions.[56] The results indicate that for a DEIS core, the consequences depend strongly on the magnitudes of the actinide release fractions. For the DCS core with Am-241 removal, the dependence on actinide release fractions is less pronounced.

Table 5: Sensitivity of consequences to actinide release fraction (cycle average).

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-M (La R.F.=0.015)					
Latent cancer fatalities	7,565	13,730	9,420	1.81	1.25
Prompt fatalities	62	88	64	1.50	1.03
ST-H (La R.F.=0.06)					
Latent cancer fatalities	13,915	32,100	18,100	2.31	1.30
Prompt fatalities	234	407	248	1.74	1.06
ST-L (La R.F.=0.003)					
Latent cancer fatalities	4,480	6,210	4,990	1.39	1.11
Prompt fatalities	26	32	27	1.23	1.04

For the DEIS core, the cycle-averaged increase in LCFs ranges from 39% to 131% for the three release fractions, corresponding to an increase of 1,730 to 18,185 LCFs for the generic site. In the worst case (ST-H), the number of *additional* LCFs is about 60% of the total number of cancers worldwide predicted to result from the Chernobyl accident. For the DCS core, the cycle-averaged LCFs are 11% to 30% greater than for LEU cores, corresponding to 510 to 4,185 additional LCFs.[57]

Results at EOC and BOC are given in Tables B.3 and B.4 of Appendix B.

Results for RG-MOX were also calculated at EOC as a function of actinide release fraction (Appendix C). Compared to an LEU core, Am-241 and Cm-242 inventories at EOC are 20 and 27 times greater, respectively. The corresponding number of LCFs is greater by 123% to 486%, approximately a factor of four greater than for WG-MOX. This significant increase in risk should be an important consideration for nations such as France and Japan that are currently using or planning to introduce RG-MOX into their nuclear plants.

In summary, we find that the number of LCFs resulting from short-term exposures following a severe reactor accident will be significantly greater for PWRs with either full or partial WG-MOX cores than for PWRs with LEU cores. In no case was the number of LCFs smaller for the WG-MOX core, unlike some of the results reported in the DOE DEIS.

The calculation of consequences due to long-term, chronic exposures following the accident is beyond the scope of this paper. Such calculations require the specification of many more parameters than calculations restricted to the period immediately after the accident and are associated with considerably greater uncertainties. However, one may expect that the cost of land decontamination will increase as a result of the greater fallout of long-lived actinides.

A Risk-Informed Analysis of MOX Loading

Compared to an LEU core, the number of latent cancer fatalities caused by a severe accident has been calculated above to be greater on average by about 85% for a DEIS core or by 22% for a DCS core. Therefore, the total risk, defined as the sum of the products of the probability and consequences for all accident sequences, will also be greater for MOX cores. In this section, the increase in risk associated with substituting MOX for LEU is estimated for these two cases.

Because the baseline risk of nuclear power plant operation is believed to be small, these risk increases are also likely to be small. To judge whether such increases will be acceptable to the NRC, however, they must be evaluated in accordance with the "risk-informed" regulatory principles that the NRC is now incorporating into its regulations. In contrast to the deterministic regulations that are now the norm, a risk-informed approach "represents a philosophy whereby risk insights are considered together with other factors to ... better focus ... regulatory attention on ... issues commensurate with their importance to public health and safety."[58]

One can assess the regulatory significance of the increased risk of MOX use by using the methodology of NRC Regulatory Guide (RG) 1.174 (see above). One of the main principles upon which RG 1.174 is based is that "when proposed changes result in an increase in ... risk, the increases should be small...".

Because the calculation of risk to the public can be very complicated, RG 1.174 defines simpler "surrogate" objectives that can be used to meet its guidelines. One of these objectives is expressed in terms of the plant LERF.

The RG 1.174 guidelines for NRC consideration of license amendment applications resulting in increases in LERF (positive ΔLERF) are as follows (RY = reactor-year):

ΔLERF < 10^{-7}/RY: application will be considered regardless of whether there is a calculation of total LERF.

10^{-7}/RY < ΔLERF < 10^{-6}/RY: application will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5}/RY.

10^{-6}/RY < ΔLERF: application would not normally be considered.

In other words, if the applicant can show that ΔLERF is below a threshold of 10^{-7}/RY, the change can be made without further review. If the proposed ΔLERF is estimated to be in the next tier, the applicant must do a more thorough calculation of the plant's baseline LERF and the expected ΔLERF to confirm the result. Finally, if ΔLERF is found to be greater than 10^{-6}/RY, the change will not ordinarily be allowed. Within each tier, the closer the result is to the upper boundary, the more intensively will the NRC review the application.[59]

At present, the use of RG 1.174 is voluntary on the part of NRC licensees. However, the NRC recently approved a staff request for the authority to use risk-informed methods to evaluate changes proposed by licensees, even if a licensee itself has not used such an approach. In reviewing license amendment requests, the staff will be able to "question the risk implications of, and potentially reject proposed changes to, the license ... where risk considerations indicate the change would be unacceptable (i.e. would not ensure adequate protection)."[60] The NRC recently used this approach to review a request by the Callaway plant to use a process known as "electrosleeve" to repair flaws in steam generator tubes.[61]

In deriving acceptance guidelines for ΔLERF from acceptance guidelines for increases in risk, RG 1.174 implicitly assumes that changes to the plant will only affect the probabilities of accidents, and not their consequences --- i.e. the source terms remain the same. However, in the case of a MOX license amendment request, the change in risk results primarily from changes in the source terms. Therefore, in order to use RG 1.174 in this case, it is necessary to modify the approach to account for changes in consequences.

To this end, one can calculate the "effective" change in LERF ($\Delta LERF_{eff}$), defined as the increase in LERF of an LEU-fueled plant that would result in the same increase in risk as the substitution of MOX for LEU with the LERF held constant. This procedure enables the ΔLERF-based acceptance guidelines in RG 1.174 to be used to evaluate the increase in risk resulting from loading of MOX.

Given the assumption that the frequencies of the severe accidents that

contribute to the LERF do not change when WG-MOX is substituted for LEU, then $\Delta \mathrm{LERF}_{\mathrm{eff}}$ can be related to the change in risk ΔR according to the following relation (Eq. 1):

$$\Delta LERF_{eff} \equiv LERF \times \Delta R / R \; = \; LERF \times (1/R)\Sigma_i (f_i \Delta C_i) \qquad (1)$$

where f_i are the frequencies of occurrence of the accidents that contribute to the LERF (i.e. $LERF = \Sigma_i f_i$), and C_i is the i^{th} "consequence coefficient," which is defined in this paper as the average risk to an individual within ten miles of a nuclear plant of contracting a fatal cancer, given the occurrence of the i^{th} plant damage state.[62] The annual LCF risk (as defined in the NRC Quantitative Health Objectives) is then given by $R = \Sigma_i (f_i C_i)$. (R also includes additional terms representing late containment failure and no containment failure, but the values of $f_i C_i$ for these plant damage states are negligible in comparison to the other terms.)

Table 6 lists values for the LCF consequence coefficients C_i for the three source terms in Table 3, averaged over a cycle. For the DEIS core, the consequence coefficients are 74% to 116% greater than those associated with an LEU core for the three source terms. For the DCS core, the coefficients are greater by 21% to 30%.

Table 6: Latent cancer consequence coefficients

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-1 (f_1=2.8x10^{-7})	0.0214	0.0372	0.0265	1.74	1.24
ST-2 (f_2=3.6x10^{-6})	0.0128	0.0276	0.0166	2.16	1.30
ST-3 (f_3=3.1x10^{-6})	0.0114	0.0208	0.0138	1.83	1.21

For the Sequoyah plant, the LERF is the sum of the three accident frequencies listed in Table 6: very early containment failure (during core degradation), containment failure at vessel breach, and containment bypass. Utilizing the frequency data and consequence coefficients given in Table 6, one finds for the DCS core that

$$LERF=7.0\times10^{-6},$$

$$R_{LEU}=8.74\times10^{-8},$$

$$R_{WG\text{-}MOX}=1.10\times10^{-7} \text{ and}$$

$$\Delta R/R=0.26.$$

Therefore, using Eq. 1, $\Delta LERF_{eff}=1.8\times10^{-6}$, which exceeds by 80% the RG 1.174 threshold of 1×10^{-6} for $\Delta LERF$. Repeating the calculation for the DEIS core, one finds that $\Delta R/R=0.99$ and $\Delta LERF_{eff}=6.9\times10^{-6}$, which is nearly seven times greater than the limit.

This calculation is based on NUREG-1150, which does not take into account external initiating events. Since external events will contribute substantially to both LERF and $\Delta LERF_{eff}$, it is necessary to include them in the analysis.

External events are analyzed in the McGuire and Catawba IPE submittals. As discussed previously, the IPEs have not been fully peer-reviewed, so these results should be considered preliminary. The total LERFs (including both internal and external events) calculated by Duke Power are 4.7×10^{-6} and 6.3×10^{-6} for McGuire and Catawba, respectively.[63] Using these values, one finds that the corresponding values of $\Delta LERF_{eff}$ for the DCS MOX plan are 1.2×10^{-6} and 1.6×10^{-6}. In both cases, the 1×10^{-6} threshold would be exceeded, and the NRC will have justification on risk grounds to prohibit the use of MOX in McGuire and Catawba, or to require significant restrictions on their licenses to lower the operating risk of the plants before MOX can be loaded. A similar conclusion is likely to apply for most other nuclear plants in the U.S.[64]

The safety case for MOX use in McGuire and Catawba appears even weaker in light of a recent SNL study finding that the Duke Power IPEs significantly underestimated the vulnerability of ice-condenser containments to early failure from hydrogen combustion during severe accidents. SNL calculates that the probability of early containment failure (given core damage) at McGuire is 13.9% for internal events, a factor of seven greater than the value estimated in the IPE.[65] Using this result, the McGuire LERF for internal events increases to 6.7×10^{-6}. If external events are considered, the total LERF will exceed 1×10^{-5}.[66] According to RG 1.174, plants with LERFs exceeding 1×10^{-5} cannot make changes that would increase the LERF by more than 1×10^{-7}. Since $\Delta LERF_{eff}$ for the DCS core is more than a factor of ten greater than this threshold, this analysis strengthens the conclusion that the use of

MOX at McGuire would pose an unacceptable risk and should not be allowed.

Therefore, unless DCS can show convincingly that the McGuire LERF is actually much smaller than the value based on recent SNL analysis, it appears unlikely that it will be permitted to load MOX in McGuire.

According to these results, the risk increase resulting from the substitution of partial or full cores of WG-MOX for LEU in most U.S. PWRs will exceed the RG 1.174 threshold. This does not imply that the NRC will prohibit MOX use, especially if it concludes that the non-quantifiable benefits of plutonium disposition outweigh the risk increase. Nonetheless, at a minimum, any utility wishing to use MOX will have to carry out a detailed level-3 PRA of its plants for both LEU and partial MOX cores, and may have significant and costly restrictions placed on the plants' operating licenses.

MOX Core Accident Probabilities

The preceding risk analysis assumed that the use of MOX fuel would have no effect on the probabilities of severe accidents. However, as discussed above, MOX fuel and LEU fuel have different neutronic and thermomechanical properties. As a result, the probabilities of severe accident sequences may increase, decrease or remain the same when WG-MOX fuel is used. The qualitative analysis below suggests that the probabilities of the most risk-significant accidents at LEU-fueled PWRs will be comparable or even higher at MOX-fueled PWRs. Therefore, the assumption that the probabilities are the same is not likely to overestimate the MOX risk and may in fact underestimate it.

Accident precursors in PWRs can be grouped into the following categories: those causing (1) an increase in power, (2) an increase in coolant temperature or pressure and (3) a decrease in coolant temperature or pressure.[67] A precursor of limiting severity in the first category is a rod cluster control assembly (RCCA) ejection; in the second, a loss-of-coolant accident (LOCA), and in the third, a main steam line break (MSLB).

Based on reactivity considerations alone, precursors in the first category may be either more or less severe in MOX cores, those in the second may be less severe and those in the third will be more severe. (However, when the thermomechanical differences of the two fuel types are taken into account, any advantages of MOX become less apparent.) The overall impact of MOX fuel use on the total core damage frequency (CDF) and LERF depends on the relative frequencies of these types of accidents, which are different for each nuclear plant. A resolution of this question will require plant-specific PRAs

for MOX cores. However, it is possible to make some general observations.

Rod cluster control assembly (RCCA) ejection

The primary coolant system in PWRs is maintained at high pressure. If a rod cluster control assembly (RCCA) were to come loose from its housing, it could be ejected from the core. This would lead to a *reactivity insertion accident* (RIA) --- a rapid increase in reactivity with local power spikes that could cause fuel damage.

Westinghouse has analyzed the RCCA ejection event for partial WG-MOX cores.[68] It is difficult to predict whether this accident would be more severe in a MOX core, because it depends on a number of competing factors, including the worth of the ejected RCCA and the delayed neutron fraction. For a partial MOX core, the worth of an individual RCCA is reduced, and if ejected, the increase in reactivity would be smaller. However, the resulting rise in power could be more rapid because the delayed neutron fraction is also smaller. According to Westinghouse, the smaller RCCA worth, along with more negative reactivity coefficients, "more than compensate for the unfavorable impact of the delayed neutron fraction."

In particular, for an RCCA ejection occurring at the end-of-cycle (at which point the burnups will also be the greatest), the maximum enthalpy was found to be 150 cal/g. Because the Westinghouse analysis uses the LEU-based criterion that fuel cladding will remain intact provided the maximum fuel rod enthalpy (absorbed heat) is below 200 cal/g, no fuel damage is predicted to occur.

However, the Westinghouse analysis does not take into account the experimental evidence from the Cabri test reactor that high-burnup MOX fuel is less resistant to reactivity insertions than LEU fuel. In fact, the calculated maximum enthalpy exceeds the 120 cal/g value at which the MOX rod rupture was observed in the Cabri test. Therefore, the increased vulnerability of high burnup MOX fuel could cancel out the beneficial effect of the reduced rod worth, unless strict burnup limits were imposed on MOX fuel.

Moreover, DCS may change the control rod material in its plants to increase the rod worths, in which case the assumption that the worth of the ejected rod would be lower in a MOX core would not be valid. Therefore, it is unlikely that this transient would be more benign for DCS cores.

Loss-of-coolant accidents

Loss-of-coolant accidents (LOCAs), initiated either directly (by pipe breaks) or indirectly (by loss of component cooling, resulting in temperature-induced failure of pump seals), rank among the most risk-significant accidents. For exam-

ple, the Catawba IPE estimates that LOCAs contribute more than 80% to the CDF for internal events. The probability that a LOCA will lead to core damage depends on the extent to which fuel cladding is damaged prior to startup of emergency core cooling, which in turn depends on the fuel rod temperatures. Although the rate of decay heat generation will be slightly lower in MOX fuel than in LEU fuel immediately after a trip (because the thermal flux is smaller), the thermal conductivity of MOX fuel is about 10% lower, and the centerline temperature about 50°C higher.

Because MOX fuel has a higher centerline temperature than LEU fuel, the rise in cladding temperature and rate of cladding oxidation during the initial stages of a LOCA can be greater than for LEU fuel, and it may be harder for MOX cores to meet the NRC's requirements for LOCA mitigation.[69]

In some transients that cause an increase in coolant temperature, such as a loss of heat sink, the stronger negative temperature feedback in MOX cores could in principle be beneficial by more rapidly forcing the reactor to a subcritical state. However, in such accidents the reactor is expected to trip automatically on excessive temperature or pressure signals, after which heat removal becomes the main concern. Therefore, the only events that may benefit from this property of MOX fuel are those in which the reactor fails to trip (the so-called "anticipated transients without scram," or ATWS), which have relatively low frequencies in PWRs. In the Catawba IPE, for example, ATWS events contributed less than 3% to the internal CDF. Therefore, a reduction in the severity of ATWS events would only have a small effect on the LERF.

Overcooling transients and pressurized thermal shock

The more negative moderator temperature coefficient in MOX cores can make overcooling events more severe, since a given rate of temperature decrease is associated with a greater rate of reactivity increase. Overcooling sequences can be initiated by many different events, most of which are associated with a leak in the secondary coolant system. This can occur either from a pipe failure, such as a main steam line break (MSLB), or as a result of a reactor trip followed by the failure of a secondary-side valve to close. Other types of events, such as steam generator tube ruptures or small-break LOCAs, can also lead to overcooling transients.

The double-ended MSLB occurring when the core is in hot standby is considered the most serious transient of this class. In this accident, the pressure and temperature of the reactor coolant system (RCS) both begin to decrease rapidly. For some core configurations, the reactor core can reach criticality and return to power. This accident is mitigated initially by an automatic high-

pressure safety injection (SI) system, which injects borated water into the core, and as the pressure continues to drop, by passive accumulators, which flood the core with more borated water.

Westinghouse analyzed the MSLB accident for both partial and full WG-MOX cores in its 1996 report. It found that even though return to criticality and peak core power occurred sooner for the MOX core, safety limits were not exceeded.

However, Westinghouse did not consider a particularly severe outcome of the MSLB accident that is likely to cause core damage and containment failure: rupture of the PWR reactor vessel as a result of pressurized thermal shock (PTS). PTS can occur if the temperature of the reactor vessel is reduced below a threshold (known as the nil-ductility transition temperature, or NDTT) while the vessel is pressurized, or if the vessel is repressurized while still cold. The risk of PTS increases with the age of the reactor vessel, because exposure of the vessel to the neutron flux in an operating reactor slowly increases the NDTT, a process known as neutron embrittlement.

To prevent the occurrence of PTS following a MSLB, two operator actions are required: termination of the auxiliary feedwater system (AFS) to the affected loop and termination of the safety injection system. The first action will halt the decrease in temperature of the RCS and the second will prevent repressurization to a potentially dangerous level.

Results of the simulation of a MSLB in a MOX-fueled PWR conducted by Westinghouse indicate that the rate of RCS temperature decrease is much more rapid in both partial and full-MOX cores than in an LEU core. After 400 seconds, the RCS average temperature in the full MOX core drops by 210°F (117°C) to 350°F (177°C), while the temperature in the LEU core drops only by around 120°F (67°C), to 440°F (227°C). According to NRC, "final temperatures of 350°F and less are expected to be potential PTS initiating events."[70] Since the MOX RCS temperature enters the region of concern sooner than the LEU core, the operator must terminate the AFS more rapidly in order to halt the decline in temperature. This decreases the likelihood that operators will be able to act in time to save the reactor vessel.

In addition, the rate of RPV embrittlement is expected to be greater in a MOX core because a greater flux of neutrons with energies greater than 1 MeV will be produced.[71] Therefore, the risk of PTS may be greater in a MOX core for two reasons --- a greater frequency of events that can overcool the reactor vessel to a dangerously low temperature, and a higher NDTT.

PTS is of special concern for Russian-designed pressurized-water reactors, or VVERs, which are especially prone to vessel embrittlement, according to the International Atomic Energy Agency. This fact should not be overlooked

as the U.S. and Russia continue to develop plans to use WG-MOX fuel in Russian VVER-1000 plants.

For a complete picture of the effect of MOX fuel on the probabilities of severe accidents, this analysis will have to be conducted in quantitative detail for each accident sequence. It is unlikely, however, that any decreases in accident probabilities that occur with MOX use will be large enough to affect the conclusions of this paper.

CONCLUSIONS

The Department of Energy's plan to dispose of ex-warhead plutonium by using it in MOX fuel in four existing commercial nuclear reactors (Catawba I & II and McGuire I & II) will impose a significant risk on the public. This paper estimates that loading 40% cores of MOX fuel containing purified WG-Pu into these reactors will result on average in a 26% increase in latent cancer risk to the public from a severe accident. This increase in risk is equivalent to that which would occur if the plant LERF --- the probability of an accident causing a large, early radiological release --- increased by the same percentage.

The NRC has the authority to examine the risk implications of a license amendment request, even if it satisfies all regulatory requirements, if it has reason to believe that the proposal would not provide adequate protection of the public health and safety. According to Regulatory Guide 1.174, a change in plant operation that increases the LERF by 25% would only be allowed for plants with a baseline LERF of $4x10^{-6}$ or less. Also, plants with LERFs exceeding $1x10^{-5}$ cannot make changes that would increase the LERF by more than $1x10^{-7}$.

PRAs conducted by Duke Power concluded that the LERFs for both the Catawba and McGuire plants exceed $4x10^{-6}$. Furthermore, a recent Sandia National Laboratories study indicates that the LERF for McGuire will exceed $1x10^{-5}$ when the vulnerability of ice-condenser containments to hydrogen explosions is taken into account. Thus the NRC will have grounds for prohibiting the use of MOX fuel in Catawba and McGuire, unless DCS can take measures to substantially reduce the baseline LERFs of these plants, or show with revised calculations that the existing risks are much smaller than indicated by their previous analyses. In either case, DCS will have to carry out peer-reviewed, MOX-specific PRAs conforming to a new quality control standard now under development.

The above risk increases were calculated under the assumption that use of MOX fuel will not increase the probability of an accident. However, this may

not be the case. Concerns with the robustness of high-burnup MOX fuel in the event of an accident have led the French nuclear safety authority DSIN to restrict the average burnup of MOX fuel assemblies to about 41 GWD/t, whereas LEU fuel is permitted to reach 52 GWD/t. The U.S. should do likewise and restrict the assembly-averaged MOX burnup to 41 GWD/t pending resolution of these safety issues. This will be a costly inconvenience for DCS, which intends at the beginning of the program to irradiate MOX fuel assemblies to 45 GWD/t, and to raise the burnup limit eventually.

The U.S. plan to encourage Russia to use WG-MOX in Russian and Ukrainian VVER-1000s poses even greater risks than the plan for U.S. domestic use of WG-MOX. VVER-1000s do not meet Western safety standards in such critical areas as fire protection and instrumentation and control systems. In addition, the pressure vessels of VVER-1000s are highly embrittled. The U.S. is not requiring that these plants be upgraded so that they fully meet Western safety standards, which would cost on the order of $150 million per unit. Given that the use of MOX will increase risk even in plants that do meet Western standards, encouraging Russia to use MOX in VVER-1000s without ensuring maximal adherence to safety is unwise.

APPENDIX A: Calculation of LEU and WG-MOX Core Inventories

This appendix contains additional information necessary to carry out the SCALE 4.3 inventory calculations described above and the results of the calculations for 58 radionuclides. Table A.1 shows the dimensions of the PWR fuel assemblies used in the calculations.

Table A.1: Fuel assembly specifications

Fuel Parameter	Value
Mass of assembly (THM)	0.423
Number of fuel rods per assembly	264
Active length (cm)	365.76
Fuel rod pitch (cm)	1.2598
Fuel rod clad outer diameter (cm)	0.9144
Fuel rod clad inner diameter (cm)	0.8001
Fuel rod outer diameter (cm)	0.7844

The fuel assembly specific powers are also needed for each fuel batch. For the EOC calculation, specific powers for each batch were approximated by the core-averaged assembly power near the midpoint of the equilibrium cycle (Table A.2). The different rate of change of reactivity with burnup of the two types of fuel results in somewhat different assembly powers and irradiation histories. BOC calculations were carried out for an exposure time of 3.4 days, corresponding to a reload fuel burnup of about 0.15 GWD/t.

Table A.2: Core irradiation histories

Batch	No. assemblies	Average power at mid-cycle (MW/t)	Average burnup (MWD/t)
LEU			
Feed	88	53.9	24,790
Once-burned	88	38.9	42,690
Twice-burned	17	15.4	49,770
Core-averaged burnup at end of cycle			35,150
WG-Pu			
Feed	92	49.6	23,860
Once-burned	92	39.6	42,910
Twice-burned	9	25.0	54,930
Core-averaged burn up at end of cycle			34,390

Full-MOX Results

Table A.3 provides the results of the calculation of radionuclide inventories at EOC for LEU and full WG-MOX cores. Fission product inventories rarely differ by more than a factor of two between the two cores. However, some radiologically important radionuclides are increased in the WG-MOX core, such as the semi-volatile isotopes of ruthenium (Ru), antimony (Sb) and tellurium (Te), while those with reduced inventories include the lower-volatility radionuclides such as yttrium (Y), strontium (Sr) and cerium (Ce).

Table A.4 lists the actinide inventories for LEU and full WG-MOX cores at the beginning-of-cycle (BOC). Because the inventories of transuranic (TRU) elements (Np, Pu, Am, Cm) are much smaller at BOC in the LEU core, while the plutonium concentration is near its maximum in the MOX core, the MOX/LEU ratios for most higher actinides are greater at BOC than at EOC.

Table A.3 shows the EOC TRU inventories for the DCS core (a 40% WG-MOX core with plutonium polishing). The significantly smaller actinide inventories are apparent.

Extrapolation to Partial MOX Cores

The radionuclide inventories for partial MOX cores can be estimated from the full-core inventories by linearly scaling the result with respect to the MOX core fraction. For instance, for a 40% MOX core (78 MOX assemblies), a given partial MOX core inventory value can be expressed in terms of the LEU and full MOX values from the equation $I_{0.4}= 55(U_0+U_1) + 37(M_0+M_1) + 5\ U_2 + 4M_2$, where U_0 is the inventory of first-cycle LEU fuel, etc. Here it is assumed that the average plutonium enrichments in MOX fuel reloads are the same for both full and partial cores.

This estimate depends on the assumption that the radionuclide inventory in a MOX assembly is insensitive to the differences in the neutron spectra of full and partial MOX cores. The validity of this assumption is evident from graphs presented in the 1996 Westinghouse study for plutonium inventories versus burnup of MOX assemblies with nearly identical fissile plutonium enrichments of 4.5% and 4.515% that are irradiated in partial and full MOX cores, respectively. Comparison of the two graphs does not reveal any visible differences in the plutonium isotopic content of the two assemblies, implying that such differences are well below 1% for all burnups. Inventories of other actinides should have a similar dependence on neutron spectrum, and inventories of fission products should be even more weakly dependent. Thus it can be concluded that the linear approximation for partial MOX core inventories is reasonable.

Table A.3: Radionuclide core inventories at EOC, DEIS Core

Fission products	LEU Core Inventory (MCi)	WG-MOX Core Inventory (MCi)	MOX/LEU Ratio
Kr-85	0.8277	0.4527	0.55
Kr-85m	24.39	14.79	0.61
Kr-87	49.04	28.42	0.58
Kr-88	68.26	37.96	0.56
Rb-86	0.1338	0.07278	0.54
Sr-89	97.49	50.21	0.52
Sr-90	7.110	3.248	0.46
Sr-91	118.9	71.48	0.60
Sr-92	125.7	84.52	0.67
Y-90	7.409	3.346	0.45
Y-91	125.6	73.07	0.58
Y-92	126.7	84.99	0.67
Y-93	95.93	71.40	0.74
Zr-95	165.6	136.2	0.82
Zr-97	153.9	142.0	0.92
Nb-95	166.1	136.1	0.82
Mo-99	177.2	172.9	0.98
Tc-99m	155.7	151.8	0.97
Ru-103	143.8	190.0	1.32
Ru-105	95.98	149.9	1.56
Ru-106	42.57	90.60	2.13
Rh-105	90.07	143.0	1.59
Sb-127	7.812	11.13	1.42
Sb-129	29.67	35.19	1.19
Te-127	7.662	11.11	1.45
Te-127m	1.240	1.974	1.59
Te-129	28.21	33.68	1.19
Te-129m	5.706	7.054	1.24
Te-131m	18.20	23.66	1.30
Te-132	135.1	139.1	1.03
I-131	93.95	99.80	1.06
I-132	137.4	142.9	1.04
I-133	194.7	191.3	0.98
I-134	216.4	205.1	0.95
I-135	185.6	183.9	0.99

Table A.3 (Continued): Radionuclide core inventories at EOC, DEIS Core

Fission Products	LEU Core Inventory (MCi)	WG-MOX Core Inventory (MCi)	MOX/LEU Ratio
Xe-133	194.8	191.7	0.98
Xe-135	45.87	77.02	1.68
Cs-134	12.47	11.99	0.96
Cs-136	3.722	6.213	1.67
Cs-137	9.422	9.331	0.99
Ba-139	172.5	159.4	0.92
Ba-140	174.0	161.7	0.93
La-140	179.8	165.6	0.92
La-141	158.0	145.5	0.92
La-142	154.5	139.1	0.90
Ce-141	159.9	147.9	0.92
Ce-143	148.5	126.4	0.85
Ce-144	115.3	90.35	0.78
Pr-143	145.7	124.0	0.85
Nd-147	63.73	60.33	0.95
Actinides			
Np-239	1755	1692	0.96
Pu-238	0.2152	0.2946	1.37
Pu-239	0.02667	0.08829	3.31
Pu-240	0.03479	0.1534	4.41
Pu-241	10.62	38.01	3.58
Am-241	0.00973	0.06633	6.84
Cm-242	2.965	21.37	7.21
Cm-244	0.1757	0.5386	3.07

Table A.4: Actinide inventories at BOC, DEIS Core

Actinides	LEU Core Inventory (MCi)	WG-MOX Core Inventory (MCi)	MOX-LEU Ratio
Np-239	984.6	966.5	0.98
Pu-238	0.06494	0.1227	1.89
Pu-239	0.01443	0.1577	10.9
Pu-240	0.01516	0.1049	6.92
Pu-241	4.664	18.43	3.95
Am-241	0.00365	0.07942	21.8
Cm-242	0.6998	8.807	12.6
Cm-244	0.03244	0.09869	3.04

Table A.5: Actinide inventories at EOC, DCS Core

Actinides	LEU Core Inventory (MCi)	DCS WG-MOX Core Inventory (MCi)	DCS-MOX/ LEU Ratio
Np-239	1755	1735	0.99
Pu-238	0.2152	0.1926	0.89
Pu-239	0.02667	0.05216	1.96
Pu-240	0.03479	0.08289	2.38
Pu-241	10.62	21.741	2.05
Am-241	0.00973	0.0245	2.52
Cm-242	2.965	6.458	2.18
Cm-244	0.1757	0.3007	1.71

APPENDIX B: Severe Accident Consequences at the End-of-Cycle (EOC) and Beginning-of-Cycle (BOC)

The results of MACCS2 calculations for severe accidents at EOC and BOC are presented in Tables B.1 through B.4. The information in these tables was used to calculate the consequences averaged over a cycle. Tables B.1 and B.2 contain the data for the three severe accident source terms in Table 3, while Tables B.3 and B.4 show the results for different actinide release fractions.

Table B.1: Consequences of severe reactor accidents at EOC

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-1					
Latent cancer fatalities	11,000	18,800	13,600	1.71	1.24
Prompt fatalities	86	129	91	1.50	1.06
ST-2					
Latent cancer fatalities	3,620	6,690	4,380	1.85	1.21
Prompt fatalities	30	40	31	1.33	1.03
ST-3					
Latent cancer fatalities	2,410	4,280	2,900	1.78	1.20
Prompt fatalities	35	34	33	0.97	0.94

Table B.2: Consequences of severe reactor accidents at BOC

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-1					
Latent cancer fatalities	4,130	8,660	5,240	2.10	1.27
Prompt fatalities	37	47	36	1.27	0.97
ST-2					
Latent cancer fatalities	1,410	3,150	1,780	2.23	1.26
Prompt fatalities	10	15	10	1.50	1.00
ST-3					
Latent cancer fatalities	1,060	2,070	1,270	1.95	1.20
Prompt fatalities	30	21	25	0.70	0.83

Table B.3: Sensitivity of EOC accident consequences to actinide release fraction

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-M (La R.F.=O,015)					
Latent cancer fatalities	11,000	18,800	13,600	1.71	1.24
Prompt fatalities	86	129	91	1.50	1.06
ST-H (La R.F.=0.06)					
Latent cancer fatalities	20,300	44,500	26,100	2.19	1.29
Prompt fatalities	359	625	391	1.74	1.09
ST-L (La R.F.=0.003)					
Latent cancer fatalities	6,410	8,680	7,140	1.35	1.11
Prompt fatalities	38	44	39	1.16	1.03

Table B.4: Sensitivity of BOC accident consequences to actinide release fraction

	LEU	WG-MOX		WG-MOX/LEU Ratio	
Source term:		DEIS core	DCS core	DEIS core	DCS core
ST-M (La R.F.=0.015)					
Latent cancer fatalities	4,130	8,660	5,240	2.10	1.27
Prompt fatalities	37	47	36	1.27	0.97
ST-H (La R.F.=0.06)					
Latent cancer fatalites	7,530	19,700	10,100	2.62	1.34
Prompt fatalities	108	188	105	1.74	0.97
ST-L (La R.F.=0.003)					
Latent cancer fatalities	2,550	3,740	2,840	1.47	1.11
Prompt fatalities	14	20	15	1.43	1.07

APPENDIX C: Consequences of Reactor-Grade MOX Use

The use of reactor-grade MOX (RG-MOX) in LWRs poses even more serious health consequences than use of WG-MOX. Because RG-MOX contains a much higher percentage of the higher-mass isotopes Pu-240, Pu-241 and Pu-242, concentrations of radiologically significant actinides such as Am and Cm increase more rapidly during irradiation of RG-MOX. Moreover, because Pu-240 and Pu-242 degrade the reactivity properties of MOX fuel, the concentration of total RG-Pu must be greater than that of WG-Pu to achieve comparable fuel energy content.

To estimate the impact of these effects, the calculations of the preceding sections were repeated for a full core of MOX containing plutonium from spent LWR fuel of 44 GWD/t burnup (see Table 2 for the initial isotopic composition of the RG-Pu). Japan is planning to construct an LWR that will use a full RG-MOX Core. The spent fuel was assumed to have been cooled for 9 years before reprocessing, and the fresh RG-MOX fuel was assumed to have been stored for three years before being loaded into a reactor. The total RG-Pu concentration of the fresh fuel necessary to achieve equivalent performance to 4.25%-enriched LEU was calculated to be 8.3%, nearly twice as great as the WG-Pu concentration. Otherwise, all other parameters were unchanged from the WG-MOX calculation. While some of these parameters, such as the average fuel assembly powers for each batch, would be different for the case of RG-MOX fuel, the differences would not be expected to have a large effect on the calculated core inventories.

Table C.1 compares the calculated actinide inventories of the RG-MOX and LEU cores.

Table C.1: RG-MOX actinide core inventory at EOC

Actinides	LEU Core Inventory (MCi)	RG-MOX Core Inventory (MCi)	MOX/LEU Ratio
Np-239	1755	1443	0.82
Pu-238	0.2152	2.667	12.4
Pu-239	0.02667	0.1368	5.13
Pu-240	0.03479	0.3532	10.2
Pu-241	10.62	86.51	8.15
Am-241	0.00973	0.2600	26.7
Cm-242	2.965	58.29	19.7
Cm-244	0.1757	3.801	21.6

Using the calculated radionuclide inventories for the end-of-cycle RG-MOX core, the MACCS2 calculations of the previous sections were repeated. Results of the consequence calculations are presented in Table C.2. Comparing Table C.2 and Table B.3, one finds that the percent increases in LCFs are about four times greater for RG-Pu than for WG-Pu.

Table C.2: Consequences of Severe Accidents Involving RG-MOX Cores at EOC

Source term:	RG-MOX	RG-MOX/LEU Ratio
ST-M		
Latent cancer fatalities	43,600	3.96
Prompt fatalities	299	3.48
ST-H		
Latent cancer fatalities	119,000	5.86
Prompt fatalities	1,590	4.42
ST-L		
Latent cancer fatalities	14,300	2.23
Prompt fatalities	59	1.55

NOTES AND REFERENCES

1. National Academy of Sciences, Committee on International Security and Arms Control, *Management and Disposition of Excess Plutonium* (Washington, DC: National Academy Press, 1994). Some observers have questioned the proliferation resistance of the proposed CIC waste form. Others have remarked that spent MOX fuel itself may not meet the spent fuel standard. The NAS is currently undertaking a study to examine these issues. It is likely that a variant of CIC will be found that clearly meets the spent fuel standard.

2. Stone and Webster filed for Chapter 11 bankruptcy in May 2000 and was acquired by the Shaw Group later that year.

3. Glenn T. Seaborg, "A Plutonium Warning From Its Discoverer," *Washington Post*, August 3, 1997.

4. At present, the greatest risk to the plutonium immobilization program is the DWPF itself. The process that was being developed to pretreat and concentrate the high-level liquid waste in the SRS tanks, "in-tank precipitation," was abandoned by DOE in 1998 because it was unable to solve the problem of excessive benzene generation. The development of a substitute technology is likely to cause substantial delays and additional costs.

5. U.S. Nuclear Regulatory Commission (NRC), Office of Public Affairs, *Probabilistic Risk Assessment,* Technical Issues Paper 12A (Washington, DC, 1999).

6. U.S. NRC, *Individual Plant Examination Program,* NUREG-1560 (Washington, DC, 1996).

7. In the introduction to NRC's IPE database, it states "The information in the database has NOT been verified or validated."

8. U.S. NRC, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,* NUREG-1150 (Washington, DC, 1990).

9. U.S. NRC, Executive Director for Operations, *Mixed-Oxide Fuel in Light-Water Reactors,* Commission Memorandum (Washington, DC, April 14, 1999), 11.

10. Ibid., 12-13.

11. This assumption may result in an underestimate of the additional consequences of using MOX fuel resulting from certain accidents, in view of recent research suggesting that high burnup MOX fuel may be prone to greater releases of both volatile and low-volatile radionuclides under accident conditions (see notes 23-27 below).

12. On an activity basis, the actinide inventory is actually smaller for MOX cores than for LEU cores, primarily because of a lower inventory of Np-239 (see Table A.1). However, the radiological hazard is not proportional to the total inventory of released radionuclides. Every radionuclide has a unique biological effect that depends on the type and energy of the radiation it emits, as well as on the manner in which it is metabolized. For instance, Np-239 is a beta-emitter, while the other actinides listed (except for Pu-241) are alpha-emitters. According to the most recent values of dose coefficients published by the International Commission on Radiological Protection (ICRP), the committed effective inhalation dose equivalent for Np-239 is 9.3×10^{-10} Sv/Bq for moderate lung absorption, compared to 4.6×10^{-5} Sv/Bq for Pu-239, 4.2×10^{-5} Sv/Bq for Am-241 and 5.2×10^{-6} Sv/Bq for Cm-242. From these values, one can show that the decrease in cancer risk associated with a 63 MCi decrease in Np-239 is only 2% of the increase associated with an additional 0.06 MCi of Pu-239 or 0.079 MCi of Am-241, and 0.06% of

the increase associated with an additional 18.4 MCi of Cm-242. Thus the risk significance of the reduction in Np-239 is small compared to the increases in the other actinides listed in the table.

13. U.S. Department of Energy, *Storage and Disposition of Weapons-Usable Fissile Materials Draft Programmatic Environmental Impact Statement*, DOE/EIS/0229-D, vol. 2 (Washington, DC, 1996), 4-690.

14. L. Devell, S. Guntay and D.A. Powers, *The Chernobyl Reactor Accident Source Term: Development of a Consensus View*, OECD/GD(96)12 (Paris: Organization for Economic Co-Operation and Development, 1995).

15. See, for example, U.S. NRC, NUREG-1150.

16. Edwin S. Lyman, Public Health Consequences of Substituting Mixed-Oxide for Uranium Fuel in Light-Water Reactors, Executive Summary (Washington, DC: Nuclear Control Institute, 1999), 5.

17. NCI briefed DOE staff on the results of its report in February 1999. In April 1999, DOE released a Supplement to the Surplus Plutonium Disposition DEIS containing revised calculations of the consequences of severe accidents at North Anna, Catawba and McGuire, based on site-specific data supplied by DCS. This analysis predicted more severe consequences from MOX accidents than had previous DOE assessments. For instance, for accidents involving early containment failure or bypass, the Supplement found that the consequences for the 40%-MOX case were from 1%-15% greater than for the LEU case, whereas the FPEIS had estimated consequences 3%-8% smaller for the full-MOX case. However, the analysis in the Supplement may need to be further revised, because DOE based its calculations on radionuclide release fractions taken from the Individual Plant Examinations (IPEs), which have not been independently validated.

18. M. Pilch, K. Bergeron and J. Gregory, *Assessment of the DCH [Direct Containment Heating] Issue for Plants with Ice Condenser Containments*, NUREG/CR-6427. SAND99-2283 (Albuquerque: Sandia National Laboratories, 2000) 110.

19. U.S. NRC Advisory Committee on Reactor Safeguards (ACRS), *Proceedings of the Meeting of the Subcommittee on Severe Accident Management* (Washington, DC, August 9, 1999).

20. Westinghouse Electric Corporation, *Implementation of Weapons-Grade MOX Fuel In Existing Pressurized Water Reactors*, DOE/SF/19683--7, report prepared for DOE (Pittsburgh; 1996) 3-1.

21. Shutdown margin is the amount of control rod worth additional to that necessary to shut down the chain reaction at the most limiting time (usually end-of-cycle).

22. Use of enriched boron may turn out not to be feasible, because it is approximately two hundred times more expensive than natural boron. However, Catawba and McGuire recycle their coolant, which could partially offset this cost.

23. J-L. Provost, Electricité de France, "Plutonium Recycling and Use of MOX Fuel in PWR: EdF Operating Experience" (paper presented at the Industry Presentation to the U.S. NRC on the Use of MOX Fuel, Rockville, MD, February 21, 1997).

24. The heterogeneity of the Cabri MOX fuel rods is characteristic of the French MOX fabrication process known as MIMAS, which Cogema has been using since 1984 and which will be the process used at the U.S. MOX fabrication plant. Although it is possible that the problem may be reduced by developing a new type of MOX fuel, DOE has

not encouraged this for the U.S. program because of the additional delays that would be needed to qualify the fuel.

25. F. Schmitz, Institute de Protection et de Surété Nucléaire, "The Status of the Cabri REP-Na Test Programme: Present Understanding and Still Pending Questions" (paper presented at the NRC/Industry Meeting on High-Burnup Fuel Issues, Rockville, MD, November 18-20, 1997).

26. A. MacLachlan, "International Meeting Fails to Resolve Questions Surrounding Cabri Future," *Nuclear Fuel*, July 27, 1998.

27. U.S. NRC ACRS, "Use of Mixed-Oxide Fuel in Commercial Nuclear Power Plants" (letter report to NRC Chairman Shirley Jackson, May 17, 1999).

28. U.S. NRC ACRS, *Proceedings of the 461st Meeting of the Advisory Committee on Reactor Safeguards*, April 9, 1999.

29. R. Clark, D. Dziadosz and S. Nesbit, "MOX Fuel Irradiation Program for Disposition of Surplus United States Plutonium," in the *Proceedings of the American Nuclear Society Embedded Topical Meeting on DOE Spent Nuclear Fuel and Fissile Material Management, June 4-8, 2000, San Diego, CA* (LaGrange Park, IL: American Nuclear Society, 2000).

30. In 1995, NRC issued a policy statement on the use of probabilistic risk assessment (PRA) which states that "the use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data..." (U.S. NRC, "Probabilistic Risk Assessment Policy Statement," *Federal Register*, 60 FR 42622, August 16, 1995). This is a departure from NRC's traditional, deterministic regulatory approach, which requires that plants demonstrate their ability to withstand a set of "design basis" accidents, which were originally developed without regard to their impact on risk, and without considering that safety systems may not work as anticipated.

31. U.S. NRC, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174 (Washington, DC, 1998).

32. U.S. NRC, "Staff Requirements Memorandum --- Proposed Guidelines for Applying Risk-Informed Decisionmaking in License Amendment Reviews," SECY-99-246 (Washington, DC, 2000).

33. Oak Ridge National Laboratory, *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, rev. 5, vol. 1, pt. 2 (Oak Ridge, TN: Oak Ridge National Laboratory, 1997).

34. U.S. DOE, Office of Fissile Materials Disposition, *Surplus Plutonium Disposition Draft Environmental Impact Statement, DOE/EIS-0283-D* (Washington, DC, 1998).

35. Westinghouse Electric Corporation, *Implementation of Weapons-Grade MOX Fuel*.

36. DOE also required that the MOX fuel should not contain integral fuel burnable absorbers (IFBAs), although Westinghouse VANTAGE-5 LEU fuel ordinarily employs IFBA coatings (a boron compound) on some fuel pellets. In order to meet this requirement, Westinghouse used wet annular burnable absorbers (WABAs) in the full-core MOX design. Because the specifications of Westinghouse WABAs are proprietary, they are not modeled in MOX core calculation below. Accordingly, IFBAs were not included in the LEU core calculation. Neither of these approximations significantly affects the results. It was observed that the presence of IFBAs would have only a small effect

(about 1%) on the core inventories. Also, a DOE study has found that the presence of WABAs does not change most MOX radionuclide inventories by more than a few percent. Among the exceptions are the plutonium inventories, which increase from 6-17%. Therefore, the MOX fuel plutonium inventory is slightly underestimated if WABAs are neglected (B. Murphy, *Characteristics of Spent Fuel from Plutonium Disposition Reactors, Vol. 3: A Westinghouse Pressurized-Water Reactor Design*, ORNL/TM-13170/V3 [Oak Ridge, TN: Oak Ridge National Laboratory, 1997], 25).

37. This occurred after the initial version of this paper was completed.

38. Although the polishing step will have the side benefit of reducing the amount of Am-241 present in the MOX fuel (thus reducing the buildup of higher actinides during irradiation), it will substantially increase the amount of liquid transuranic (TRU) waste from the MOX fabrication process. DCS has estimated that the MOX fuel fabrication facility, including the polishing module, will generate 500 liters of liquid TRU waste per year, a factor 1000 times greater than DOE's estimate. This waste stream is almost entirely a result of the polishing step (U.S. DOE, *Supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement*, Appendix P [Washington, DC, 1999], 8.)

39. Diablo Canyon Nuclear Power Plant, Units 1 & 2, *Final Safety Analysis Report Update*, November 11, 1996.

40. Some of the other data from the 1996 Westinghouse report has been modified. Unlike the Pu in the Westinghouse report, the Pu in the DEIS contains Pu-238 and Am-241 and has a slightly lower fissile content. For this reason, the Pu concentration in the fuel was increased slightly in order to achieve equivalent performance, from 4.37% Pu to 4.54% Pu+Am (4.50% Pu). This value was determined by calculating the Pu-239 equivalent for both compositions (Nuclear Energy Agency, *Plutonium Fuel: An Assessment*, Annex G [Paris: Organization for Economic Cooperation and Development, 1989], 120). For plutonium that has been polished and the initial Am-241 removed, the corresponding concentration is 4.43%. Also, the Westinghouse MOX core design uses fuel assemblies with two different Pu concentrations. For simplicity, the calculation below assumes that the Pu concentration of all assemblies is equal to the average concentration in the reload fuel. The actinide inventories calculated using this approximation agree with the Westinghouse report to better than 1%.

41. The MOX core has a larger reload size because the reactivity, temperature and fission gas generation of MOX fuel are greater than those of LEU fuel at high burnups. Increasing the size of the reload results in a smaller number of MOX assemblies in the third cycle. However, this slight difference is not important for the calculation.

42. The DCS core calculations assumed that both LEU and MOX fuel assemblies will be irradiated for three cycles, consistent with the long-term plan of DCS. However, since DCS has said that it will irradiate MOX fuel for only two cycles initially, this case was also studied. Because only four MOX fuel assemblies are present in the third cycle, the results are nearly identical for the two cases.

43. U.S. DOE, *Surplus Plutonium Disposition DEIS*, vol. 2 (Washington, DC, 1998), K-6. This paper assumes that a period of thirty years elapses between the original fabrication of the Pu pit and the loading of the corresponding MOX fuel into the reactor, whereas DOE assumes an elapsed time of sixty years. The former assumption is based on the fact that pit refabrication operations at Rocky Flats ceased in the late 1980's, and DOE plans to load MOX fuel into reactors from 2007 to 2022. This difference is only important for the DEIS case where the plutonium is not “polished” (removing

most Am-241) prior to fuel fabrication.

44. The concentration of neptunium-239 (Np-239) was found to be 4% smaller in the DEIS case but only 1% smaller in the DCS case. Pu-238, which is produced both by repeated neutron absorption in U-235 and by decay of Cm-242, is larger for the DEIS case but smaller for the DCS case.

45. D.I. Chanin and M.L. Young, *Code Manual for MACCS2: Volume 1, User's Guide*, SAND97-0594, (Albuqurque, NM: Sandia National Laboratories, 1997). While preparing this paper, the author discovered an error in the MACCS2 software which resulted in the overcounting of cancer fatalities among individuals receiving committed effective doses (CEDs) greater than 10 Sv and a consequent overestimate of cancer risk. While the error will not be fixed until release of the next version of MACCS, an "unofficial" corrected version of the code was provided to the author. All data in this report has been generated with the corrected, unofficial version of MACCS2. After the author pointed the error out to DOE, it also revised its calculations for the SPD final EIS.

46. The last indicator is one of the risk measures specified in the Quantitative Health Objectives (QHOs) of the NRC's 1986 Safety Goal Policy Statement (U.S. NRC, "Safety Goals for the Operations of Nuclear Power Plants: Policy Statement," *Federal Register*, 51 FR 30028, August 4, 1986). The QHOs state that the risks of prompt and latent cancer fatalities to individuals in the vicinity of a nuclear plant boundary should not exceed 1/1000 of the risks of these conditions resulting from all other causes. "Vicinity" is defined as within one mile of the plant boundary for PFs and within ten miles for LCFs. According to the NRC, these annual limits are approximately $5x10^{-7}$ and $2x10^{-6}$ for PFs and LCFs, respectively.

47. International Commission on Radiological Protection (ICRP), *Age-Dependent Doses to Members of the Public from Intake of Radionuclides: Part 5, Compilation of Ingestion and Inhalation Dose Coefficients*, ICRP Publication 72 (Oxford: Pergamon Press, 1996).

48. The MACCS2 code converts radiological exposure to radiation dose using coefficients issued by the Environmental Protection Agency (EPA). These are based on outdated values issued in 1979 by the International Commission on Radiological Protection (ICRP) in Publication 30. The revised values in ICRP 72 will not be officially adopted in the U.S. for many years, but they were used in this paper because of their potential to affect the results. Interestingly, although the absolute numbers of latent cancers decreased by about one-third after the data was changed, the MOX/LEU ratios did not change by more than a few percent, and even increased in some cases.

49. MACCS2 divides the cancer risk by a factor of 2 (the "dose and dose-rate effectiveness factor," or DDREF) for individuals who receive doses of less than 0.2 Sv. While this may be appropriate for low linear energy transfer (low-LET) radiation, such as gamma and beta radiation, there is little evidence that such a factor is appropriate for high-LET radiation from internal alpha-emitters. In fact, there is evidence that the effectiveness of high-LET radiation in inducing cancer is actually enhanced at low dose rates (A.C. Upton, "The Linear Non-Threshold Dose-Response Hypothesis: A Critical Reevaluation," in *Proceedings of the 26th NRC Water Reactor Safety Information Meeting*, NUREG/CP-0166, vol. 1 [Washington, DC: U.S. NRC, 1999], 189). Therefore, the impact of the release of alpha-emitting actinides on the number of LCFs may be underestimated by MACCS2.

50. The MACCS2 calculations were carried out for a reactor situated in a generic region with a constant population density of 100 persons per square kilometer, no rain,

a wind speed of 2 m/s and Pasquill class D atmospheric conditions. Evacuation of the population within 10 miles was assumed, triggered by a warning issued at containment breach.

51. This categorization is an oversimplification and can conceal differences in radionuclide properties which may be important. For instance, the vapor pressure of americium metal is thousands of times greater than that of lanthanum metal at typical severe accident temperatures, which suggests that under reducing conditions (i.e. high hydrogen concentration), these two radionuclides could behave quite differently. However, accounting for the specific properties of every element in a reactor core would substantially increase the complexity of risk calculations.

52. R. Davis, A. Hanson, V. Mubayi and H. Nourbakhsh, *Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants*, NUREG/CR-6295 (Washington, DC: U.S. NRC, 1997), 3-21.

53. Current nuclear plant PRAs are generally considered proprietary and are unavailable to the public. Some older PRA information for these plants is available from their IPEs (i.e. Ref. 63). However, the results of the IPEs have not been verified by NRC staff or other independent reviewers. In addition, the IPEs contain numerous anomalies which call their accuracy into question, especially for Duke Power's nuclear plants. For instance, a recent NRC analysis has shown that ice-condenser containments, and McGuire in particular, are more vulnerable to early containment failure than is indicated by Duke Power's IPEs.

54. The results of consequence calculations like these are highly uncertain, largely because of uncertainties in the underlying source terms. However, the increase in consequences from MOX use was observed over a wide range of actinide release fractions, and it can be reasonably concluded that this represents a clear trend of increased accident consequences for MOX cores.

55. While the use of partial MOX cores has a lower annual risk to the public than use of full cores, the disposition program will take longer to carry out, distributing the total risk over a longer period. Since the increase in risk is approximately proportional to the MOX core fraction, the total risk to the public will be the same in both cases.

56. An NRC analysis of actinide release fraction uncertainty distributions finds that the mean values of the distributions are approximately 1% of the core inventory, the upper bound is about five times greater than the mean, and the 75% percentile is approximately five times smaller than the mean (U.S. NRC, *Accident Source Terms for Light-Water Nuclear Power Plants*, NUREG-1465 [Washington, DC, 1995], 33). For strontium, the upper bounds are estimated to be about three times greater, and the 75% percentile values five times smaller, than the mean values. ST-H was obtained by multiplying the low-volatile release fractions in Table 3 by a factor of five, without allowing them to exceed the NRC estimates of their upper bound values (6% for La and Ce, 41% for Sr). ST-L was obtained by dividing the mean values for all low-volatile groups by a factor of 5.

57. To assess the impact of MOX burnup on these results, the case for which MOX fuel is only irradiated for two (instead of three) cycles was also evaluated. The results were only lower by a few percent.

58. U.S. NRC, *White Paper on Risk-Informed, Performance-Based Regulation* (Washington, DC, undated).

59. NRC emphasizes that the guidelines in RG 1.174 are “not to be used prescrip-

tively." For example, an application associated with an unacceptable ΔLERF may be considered if "it is shown that there are unquantified benefits that are not reflected in the quantitative risk results." DCS may well argue that this caveat applies to plutonium disposition. However, the fact that there are safer disposition alternatives would also have to be considered.

60. U.S. NRC, *Options for Risk-Informed Revisions to 10 CFR Part 50 --- "Domestic Licensing of Production and Utilization Facilities,"* SECY-98-300 (Washington, DC, 1998).

61. U.S. NRC, *Electrosleeve Amendment Issued to Union Electric Company for Callaway Plant, Unit 1*, SECY-99-199 (Washington, DC, 1999). In this case, a new technique for repairing cracked steam generator tubes was questioned because the repaired tubes would be prone to failure when exposed to the high temperatures that would occur during a severe loss-of-coolant accident, even though they were shown to be resistant to failure at the lower temperatures that would occur during a design-basis accident.

62. RG 1.174 states that the restrictions on LERF are surrogates for the PF QHO, which is considered more restrictive than the LCF QHO (see note 46). This is true in an absolute sense, since the PF QHO is smaller than the LCF QHO. However, for the case of MOX loading, the changes in LCF risk are greater than the changes in PF risk. Therefore, the restriction on ΔLERF is more conservative when applied to the change in LCFs.

63. Duke Power Company, *McGuire Nuclear Station IPE Submittal Report*, November 1991; *Catawba Nuclear Station IPE Submittal Report*, September 1992.

64. For the much larger group of PWRs with large dry containments, like the North Anna plant (until this year part of the DCS consortium), the average IPE frequencies (for internal events only) of early containment failure and containment bypass were each found to be $5x10^{-6}$, and the average LERF for all PWRs was found to be $9x10^{-6}$. When external events are included, the average total LERF is likely to be well over $1x10^{-5}$. One can then conclude that most PWRs with large dry containments will also violate the RG 1.174 limit for partial or full MOX core loadings. While BWRs have not been explicitly considered in this article, the results should be very similar to those for PWRs. In fact, the maximum actinide release fraction for BWR accidents identified by NRC is 11%, nearly twice as great as the PWR value. Since the average internal LERF for U.S. BWRs is about $5x10^{-6}$, $\Delta LERF_{eff}$ values for BWRs should be similar to those for PWRs.

65. Pilch, Bergeron and Gregory, *Assessment of DCH for Ice Condensers,* xviii.

66. The IPE core damage frequency (CDF) for internal events for McGuire is $4.0x10^{-5}$. SNL's estimate for the early containment failure frequency is 13.9% of this value, or $5.6x10^{-6}$, compared to the IPE estimate of $8.2x10^{-7}$. Adding the IPE frequencies of containment bypass ($9.6x10^{-7}$) and containment isolation failure ($1.3x10^{-7}$) yields a total LERF for internal events of $6.7x10^{-6}$. If one assumes that the same early containment failure conditional probability of 13.9% also applies to external events, a similar calculation yields a LERF for external events of $5.0x10^{-6}$ and a total LERF of $1.2x10^{-5}$. However, this assumption will underestimate the risk for external events, because external events are more likely to result in early containment failure than internal events, particularly for ice-condenser plants. This is because external events are more likely to occur with a loss of off-site power, which almost inevitably would lead to early containment failure from a hydrogen explosion, due to the loss of AC-powered hydrogen

igniter systems (Pilch, Gregory and Bergeron, *Assessment of DCH for Ice-Condenser Plants*, 113).

67. U.S. NRC, *Final Generic Environmental Impact Statement on the Use of Recycle Plutonium in Mixed-Oxide Fuel in Light Water Cooled Reactors* (GESMO), NUREG-0002 (Washington, DC, 1975), IV C-61.

68. Westinghouse claims also to have done the analysis for full MOX cores, but a careful reading of their report indicates that they have not actually done so.

69. U.S. NRC, *Mixed-Oxide Fuel*.

70. U.S. NRC, *Pressurized Thermal Shock (PTS)*, SECY-82-464 (Washington, DC, 1982), G-1.

71. DCS plans to limit the fast neutron fluence on the reactor vessel by using a "low-leakage pattern," in which fresh MOX fuel assemblies will not be loaded in positions near the core periphery. However, this scheme can increase MOX power peaking and has caused difficulties in at least one European plant using MOX. It is not clear if both problems can be resolved adequately at the same time.