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ORNL/LTR-2012/315

The Use of MOX Fuel in the United States: Bibliography of Important Documents and Discussion of Key Issues

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Prepared for the
Department of Energy
Office of Fissile Materials Disposition

UT-BATTELLE
ORNL-27 (4-00)

ORNL/LTR-2012/315

Reactor and Nuclear Systems Division

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managed by
UT-BATTELLE, LLC
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

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1. INTRODUCTION

The United States and Russia have agreed to each dispose of no less than 34 metric tons (MT) of plutonium withdrawn from nuclear weapons programs. The U.S. implementation program includes conversion of weapons-grade (WG) plutonium into nuclear fuel (known as mixed-oxide or MOX fuel) and irradiation of the MOX fuel in commercial nuclear power plants. The MOX fuel will be substituted for a portion of the low-enriched uranium (LEU) fuel that these reactors currently use.

From the 1960s to the 1980s, significant amounts of MOX testing were performed at various reactors in the United States. Reactor grade as well as near weapons grade MOX had already been irradiated and tested in numerous test and commercial reactors in 1960s and 1970s. Recent activities aside, the last MOX assemblies irradiated in U. S. plants were irradiated in the Ginna reactor and discharged in 1985. MOX fuel with isotopics near WG MOX (a $^{240}Pu/^{239}Pu$ ratio of 0.115) was used in the Quad Cities MOX testing (1970s). In the Saxton program (irradiation of nuclear fuel in a test reactor), 91.4% fissile ($^{239}Pu+^{241}Pu$) plutonium was tested. Considerable core fractions of MOX were also used in the Big Rock Point reactor. Much of the U.S. work ultimately culminated in a *Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in LWRs* (NUREG-0002) issued by the U.S. Nuclear Regulatory Commission (NRC).

From the 1970s, widespread industrialization of reactor-grade (RG) MOX occurred in several European countries as well as in Japan. France, Belgium, Germany, Switzerland, and Japan have all licensed MOX fuel for use in LWRs (Blanpain et al., 2004). In Europe and Japan, 40 reactors use or have used RG MOX fuel. Of the 36 MOX-using reactors in Europe, 34 were PWRs and two were BWRs. Until recently, Germany had 10 reactors using MOX fuel. Switzerland uses MOX fuel in three reactors; France uses MOX in 21 reactors, and Belgium uses MOX in two reactors. As of May 2008, 4598 MOX PWR fuel assemblies and 804 MOX BWR fuel assemblies that were fabricated by AREVA NP and the companies it acquired have been irradiated in Europe (Areva Doc No 12-9123377-000, September 2009). MOX fuel assembly discharge burnup exceeds 50 GWd/MT for BWR and 60 GWd/MT for PWR fuel without any penalty on core operating conditions or fuel reliability (Blanpain et al., 2011). Government, vendor, and utility sponsored scoping studies and comprehensive assessments covering the in-core performance of MOX as well as the reactor operational and accident response have been performed worldwide. The overall widespread success of the industrial use of MOX greatly reduced technical uncertainties regarding the decision to proceed with MOX as a major disposition option for WG plutonium. This history is summarized in ORNL/TM-13428. A relatively recent look at the Quad Cities irradiations is summarized in ORNL/TM-13567.

Activities concerning the use of MOX were restarted in the United States in the mid-90s when the Department of Energy (DOE) and the National Academy of Sciences explored the feasibility of irradiating WG-Pu as MOX fuel to eliminate surplus weapons plutonium. A number of evaluations, analyses, and tests concerning how WG-MOX fuel may perform and how it differs from either reactor-grade MOX or LEU fuel were conducted and involved a wide variety of reactors (PWRs, BWRs and HWRs). Numerous government, vendor, and utility sponsored scoping studies and comprehensive assessments covering the in-core performance of WG-MOX as well as the reactor operational and accident response have been performed in the United States and internationally. Using this information, the *Surplus Plutonium Disposition Final Environmental Impact Statement* (U.S. DOE, DOE/EIS-0283) was issued. The evaluations also included the completion of testing of WG-MOX test rods in the Advanced Test Reactor (ATR) in Idaho and subsequent post-irradiation examinations of the test rods at ORNL (Hodge et al., 2005; Hodge and Ott, 2006). That evaluation was performed primarily as a generic test for gallium impurities in the plutonium, which could be a difference between the use of weapons-derived material and the historical and worldwide experience with MOX fuel. After down-selection of a

MOX fuel supplier/utility consortium (at that time—Duke Cogema, Stone & Webster), the irradiation of MOX lead test assemblies (LTAs) in the Duke Catawba Unit 1 PWR during cycles 16 and 17 was performed. Initially, these LTAs were scheduled for three cycles of irradiation but they were withdrawn after two cycles due to higher than expected assembly axial growth and rod bow. The two cycles of irradiation bounded the planned burnup for MOX fuel; the third cycle was intended to support a future increase in the MOX fuel burnup limit, if needed. The additional growth was attributed to a relatively new guide tube material that was being utilized in the AREVA LEU assembly design; neither the growth nor rod bow were considered to be a MOX fuel issue. Five of the MOX fuel rods from the Catawba MOX LTAs were shipped to Oak Ridge National Laboratory for post-irradiation examination (McCoy et al., 2010; McCoy et al., 2011).

Besides DOE and nuclear power vendors, a number of other groups such as the organizations International Atomic Energy Agency (IAEA), the Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) Task Force on Reactor-based Plutonium Disposition (TFRPD), national and international academic institutions, and public interest groups have also researched the technical, economic, and policy issues associated with the use of MOX fuel in commercial LWRs in the United States. These independent assessments vary widely in their methodology, assumptions, data quality, and computational tools and models. Thus, their results vary widely and conclusions drawn from the studies may be difficult to interpret without having a clear understanding of the inputs used in developing the reports, making it challenging to compare assessment results between organizations.

Recognizing both the extended timeframe over which these studies have been conducted and the potential for conflicting conclusions, this paper was developed to provide an annotated bibliography of many of the most relevant publically available reports that relate to the use of WG-MOX fuel in U.S. reactors. The annotated bibliography of select documents as well as a more complete list of relevant documents has been compiled covering a range of topics, such as MOX fuel concepts, core design, fuel cycle assessments and options, operational behavior and experience, code benchmark studies, severe accident consequences and models, and public opinion reports. This bibliography can be found in Appendix A. Copies of many of the documents listed are available at ORNL.

This paper also summarizes the general consensus reached in these reports on some of the more relevant issues related to WG-MOX fuel. However, to better understand these issues, the data used, uncertainties inherent in any calculations, and key assumptions made by the authors, the referenced documents should be read in their entirety.

Although there are many issues discussed in the referenced documents, several have garnered special attention. Section 2 discusses general similarities and differences between WG-MOX and LEU. Section 3 covers severe accident differences between MOX and LEU cores. A summary is provided Section 4.

It should also be recognized that the U.S. Nuclear Regulatory Commission (NRC) will be licensing WG-MOX fuel based not only on the publicly available information cited in this and similar documents, but also on detailed, WG-MOX specific information provided them by the fuel vendors. That information, although proprietary in nature will include detailed experimental and operational data, updated and validated modeling information, and other documentation required by NRC regulations to license new fuel types. NRC will then issue utility specific, publicly available Safety Evaluation Reports on their evaluation of the suitability of WG-MOX fuel for use in LWRs.

2. SIMILARITIES AND DIFFERENCES BETWEEN WG-MOX AND LEU—NORMAL OPERATIONS

In practice, MOX fuel and LEU fuel perform similarly in reactors during normal operations. Apart from the fuel pellet material, MOX fuel assemblies and LEU fuel assemblies are similar with respect to thermal-hydraulic and mechanical design and fabrication (IAEA 2003; IAEA 2011). Accommodations for fuel zoning in MOX assemblies must be made to compensate for flux gradients from adjacent LEU assemblies. In addition, co-resident LEU fuel may be slightly redesigned (with enrichment grading, for example) to better accompany the partial MOX core. Both MOX fuel rods and LEU fuel rods consist of sintered ceramic pellets that are predominantly ^{238}U dioxide (UO_2), and the respective material properties are similar (IAEA, 2003; Fujishiro et al., 1999). The microstructures of the two types of fuel pellets differ somewhat, in that LEU fuel is a homogeneous mixture of ^{238}U and ^{235}U dioxide, while MOX fuel (for master-mix fuels) is more heterogeneous, with very small plutonium oxide rich agglomerates in a matrix of depleted uranium dioxide (^{238}U). The nuclear characteristics of MOX and LEU fuel are also different, due primarily to the nuclear cross-section differences between uranium and plutonium. However, the MOX fuel assembly neutronic design is adjusted to make the MOX fuel nuclear characteristics compatible with co-resident LEU fuel; therefore, the resulting fuel performance is similar. It should also be noted that LEU fuel derives a third or more of its energy from the in-core Pu production that occurs during LEU in-core irradiation (Graves, 1979), so the differences between LEU and MOX fuel decrease with increased burnup.

The differences between MOX and LEU fuel include:

- The thermal fission and overall absorption cross sections of ^{239}Pu are substantially higher than those of ^{235}U. Thus, for the same power level, MOX fuel has a lower thermal neutron flux. This leads to a reduction in the effectiveness of thermal neutron absorbers in a partial MOX fuel core, most notably in soluble boron, burnable absorbers, and control rods. This effect may be addressed by various means, including one or more of the following methods: increasing soluble boron concentration, using enriched soluble boron, adding additional control rods to reactors with partial MOX fuel cores, or by developing a core design to ensure adequate shutdown margin. (Gehin, Carbajo, and Ellis, 2004).
- The flux gradient between LEU fuel assemblies and MOX fuel assemblies requires pressurized water reactor MOX fuel to incorporate low plutonium concentration zones on the exterior of the fuel assembly. Otherwise, those exterior MOX fuel rods would experience unacceptably high peaking due to thermal neutrons leaking in from the adjacent LEU assemblies with higher neutron flux levels. The intra-assembly zoning for MOX fuel assemblies successfully minimizes the power peaking that would otherwise be experienced in partial MOX fuel cores. (IAEA, 2003)
- Fission gas release from MOX fuel at elevated burnups (greater than 40 GWD/t) is higher than the fission gas release from LEU fuel. This effect has been predominantly tied to the relatively higher power experienced by MOX fuel at high burnups. The higher fission gas release has been successfully addressed in France and Belgium by modifying the MOX fuel rod design to provide more plenum space.
- Greater capture-to-fission ratios for MOX fuel result in the production of fissile ^{241}Pu, which in turn leads to a slower decrease in reactivity for MOX as fuel burnup increases.
- The radionuclide inventory in spent MOX fuel differs somewhat from that of spent LEU fuel. The distribution of fission product isotopes is slightly different, and the irradiated (spent) MOX fuel contains more plutonium and minor actinides.
- As a result of different fission product and actinide inventories, the decay heat from MOX fuel is slightly lower than that of LEU fuel immediately following shutdown, providing a safety benefit during the timeframe of most concern for analyses of postulated transients and accidents. In the longer term, the decay heat from MOX fuel exceeds that of LEU fuel, and that difference must be

taken into account for spent fuel management. (Gehin, Carbajo, and Ellis, 2004; IAEA, 2011; Ade and Gauld, 2011)

- Because of the increased Pu-content, MOX-fueled nuclear reactors have a lower delayed neutron fraction and shorter neutron lifetimes than LEU-fueled nuclear reactors. This can lead to faster reactor core responses to reactivity transients. (Sowder, A., 2009; Gehin, Carbajo, and Ellis, 2004).

The isotopic composition of plutonium used in MOX fuel (Table 1) depends on the manner in which it was produced (reactor design, initial uranium enrichment, fuel discharge burnup, and spent fuel storage time). Generally, reactor grade Pu refers to recycled Pu obtained from spent LEU fuel that has been irradiated in an LWR for an extended time and contains significant amounts of Pu isotopes other than ^{239}Pu. Weapons grade Pu is derived from very low burnup uranium to optimize the fissile ^{239}Pu content while minimizing the other Pu isotopics (IAEA, 2003; Trellue, 2006). Table 1 compares typical ranges of isotopic concentrations for WG and RG plutonium. Once Pu is formed in an LWR LEU core, neutron capture and decay reduce the fissile Pu fraction relative to total plutonium with time (Sowder, A., 2009).

Table 1. Plutonium isotopic composition of weapons and reactor-grade MOX
(Duke et al.; 2002; IAEA, 2003; Trellue, 2006; Duke et al., 2001; Kang et al., 2000; Sowder, 2009)

Isotope	Plutonium grade	
	Weapons grade (wt %)	Reactor grade (wt %)
^{238}Pu	0	1–4
^{239}Pu	92–95	50–60
^{240}Pu	5–7	24–27
^{241}Pu	0–0.5	6–11
^{242}Pu	0–0.05	5–10

3. ACCIDENT PROBABILITY, CONSEQUENCE, AND RISK

3.1 Background[1]

A probabilistic risk assessment (PRA) is used to estimate off normal occurrences in a nuclear power plant, how likely an accident may be, and what the consequences may be for each accident. A Level 1 PRA estimates the frequency of accidents that lead to reactor core damage—the core damage frequency (CDF). This is done by putting together event tree models that represent sequences of occurrences that could lead to core damage. The frequencies for all core damage sequence elements are combined to calculate the total core damage frequency. A Level 2 PRA models the plant's response to the Level 1 PRA accident sequences that resulted in reactor core damage. Such core damage sequences are typically referred to as ***severe accidents***. Toward that end, a Level 2 PRA analyzes accident progression by considering how the containment structures and systems respond to the accident. Thus, the analysis is plant specific. A Level 2 PRA takes in to consideration the accident end states and the key phenomena that affect accident progression and its resultant source term (source term includes the quantity, timing,

[1] Much of this information is taken from http://www.nrc.gov/about-nrc/regulatory/risk-informed/pra.html#Level1.

and energetics of the radioactive material release). Thus, the PRA takes into account the complicated phenomena of an accident and produces best estimate source terms for the events considered. A Level 3 PRA is the consequence analysis of the accident. In this analysis, the source terms from a Level 2 PRA are put in to a consequence code along with site specific weather (usually a full year of data), population sector and evaluation information, and a number of other atmospheric and dosimetric assumptions. Health effects and other consequences from the accident can then be calculated as a function of the weather scenarios at the time of the accident (by weather sampling). Subsequent to this, the analysis may proceed to develop risk integration which assembles the probabilities of the accidents, the source terms, and accident consequences into overall expressions of risk.

3.2 PRAs, Source Terms and Public Risk

The conditions leading to low probability, high consequence accidents are considered and evaluated during a nuclear plant's design evaluation, during its licensing process, and in the plant's probabilistic risk evaluation. In severe accident scenarios involving core melt and release of high consequence source terms (e.g., low probability/high consequence accidents), it is important to realize that there are many phenomena that the safety analyst must model to calculate risk (probability of the accident times the consequence).

Many assumptions have a significant impact on the calculation of an accident source term. A treatise of all of the phenomena and all of the assumptions that must be evaluated in such an analysis is beyond the scope of this paper. The percentage of the core that could be released is difficult to determine with a reasonable degree of certainty. Release fraction models have been a multi-decade, long-term research area for the DOE, utilities, the NRC, and the international community. Similarly, modeling the atmospheric dispersion and the accurate estimation of consequences to human health from a release presents challenges.

Cores with partial MOX fuel assemblies have greater inventories of some radioactive isotopes (e.g., plutonium) and lesser inventories of other isotopes (dependent on specific fuel design details, characteristics of each fuel cycle, and fuel burnup levels). Government, industry, and nongovernment entities have studied differences in the expected risk of reactor accidents from the use of MOX fuel. It has been suggested that the substitution of WG-MOX for LEU fuel will affect both the probability of accident occurrence and the consequences of each reactor accident sequence. In general, releases of radioactive isotopes of cesium and iodine account for the largest potential exposures to the public—significantly larger than those expected from the actinides; thus, differences between MOX fuel and LEU fuel in severe accidents may be small compared to the uncertainties in the analyses themselves. Depending on the source term and situation modeled, differences less than roughly an order of magnitude may be considered indistinguishable due to the uncertainty inherent in the analyses.[2]

Uncertainty analysis on important accident parameters and how the uncertainties propagate and impact risk is also frequently evaluated. There are uncertainties at every stage of a PRA. Additionally, source terms are plant specific. A presentation by H. P. Nourbakhsh and T. S. Kress, *Assessment of Phenomenological Uncertainties in Level 2 PRAs* (Nourbakhsh, P., November 2005), may be of help to the reader. Uncertainties were also analyzed during the NUREG-1150 PRA that the NRC conducted in the 1990s. In the NUREG-1150 study, panels were used to examine uncertainties if they were large and important to risk (see Table A.1, page A-41 of NUREG-1150, Volume 1). Examination of this table

[2] An informal elicitation of nine meteorology experts on short-range dispersal coefficients on a sample application resulted in dispersal values that ranged over about an order of magnitude. (N. Bixler, R. Waters, and D. Whitehead, Sandia National Laboratories, *Uncertainty Analysis with MACCS2 Using Data Based on Expert Elicitation*, American Meteorological Society Annual Meeting, 2002).

shows the phenomena of most concern. The complicated phenomena in this table can have major impacts on the source term and risk. (NUREG/CR-4551 is a useful reference on this topic).

Phenomena regarding core melt progression and the resulting fission product releases are very complex. Over the past 25 years, large computer models such as MELCOR and MAAP have been developed and continuously upgraded to provide a tool for evaluating severe accidents and estimating source term releases. For severe accidents, it has been suggested that accident progressions for both LEU and partial MOX cores could be the same or very similar. Specific MELCOR computer analyses on LEU and MOX cores have been performed to verify this assumption for PWRs (performed on Catawba and McGuire—see Ashbaugh et al., 2010). These studies considered specific fission product release tests performed on MOX as well as the difference in decay heat and found them to not be significantly different (see Fig. 2.7, Ashbaugh et al., 2010). Accident progression and timing were found to be similar between the two core types, and any differences in source terms were thought to be within the uncertainties of the computer models.

To put source terms in perspective, it should be noted that a recent analysis was documented in SAND2008-6664 (Asbaugh et al., 2010) regarding the NUREG-1465 (published in 1995) release fractions. The new analyses suggested that the NUREG-1465 release fractions are conservative (too high) by about a factor of 2 and that release durations are longer than the NUREG-1465 durations. It was stated that more analysis was needed prior to making modifications to the NUREG source term. This indicates that as more data becomes available to researchers, predictions of source terms can change significantly. This is due to the complexity involved in the phenomena that are being modeled and the incremental availability of experimental information that is factored into such efforts. *When viewed with this understanding, inventory differences and their resulting changes in source terms between LEU cores and partial MOX cores do not appear to be significant.*

3.3 Consequence Analysis—MOX and LEU

Given a spectrum of source terms, the next step in assessing overall risk is to perform a consequence analysis. Figure 1 below shows the major dose pathways and phenomena typically considered in a consequence analysis.

When developing a consequence analysis, the uncertainty associated with each parameter should be characterized. There are two major classes of uncertainty in a consequence analysis. The first is called "stochastic" uncertainty. This is the uncertainty that is inherent in a physical process (such as uncertainty in knowing the actual weather at the time of the hypothetical accident), and this uncertainty cannot be reduced because it is by definition, unknown. Since we do not know when an accident will occur, or the type of weather that will be present at the time of a severe accident, consequences of reactor accidents in Level 3 PRAs are done using sampling for weather-related parameters to quantify this uncertainty's impact on the analysis results. As an example, the direction and speed of the wind, along with precipitation and the downwind exposed population greatly impact the calculated health effects of any radioactive release. A single weather scenario says very little about the actual risk. For some severe accidents, the greatest impact may be a weather scenario where a plume of aerosols is carried downwind to a population where a rain event then washes out the aerosols onto the population. Obviously, analysis of a dry weather scenario would not capture the consequence aspects of that scenario. The solution is to have a model that captures the high probability site-specific weather scenarios. As an example, Fig. 2 below shows the distribution of consequences (varying with weather) assuming a specific source term at the Surry plant. It's important to note that the consequence alone can vary by over 2 orders of magnitude based on the weather uncertainty alone.

To capture the stochastic uncertainty, the results of analyses are often plotted on a curve called a complementary cumulative distribution function (CCDF). A CCDF curve plots the probability of exceeding a specific consequence (such as prompt fatalities or latent cancers per reactor year) as a

function of that consequence (number of prompt fatalities or latent cancers) using 1 years' worth of weather data. This is simply an alternative form of displaying the information in Fig. 2.

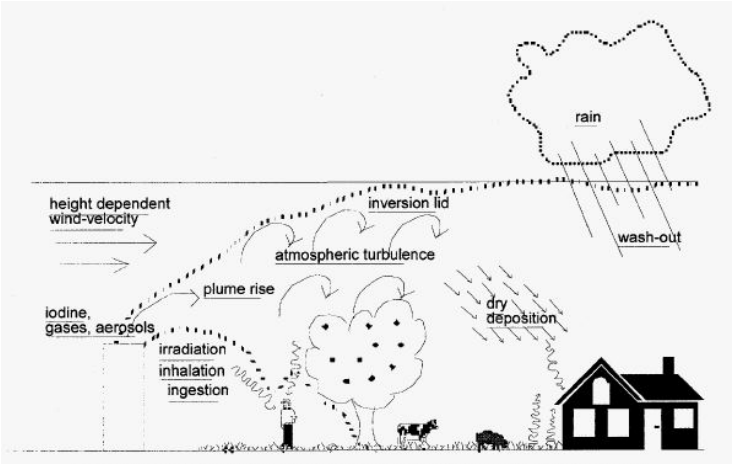

Fig. 1. Dispersion and deposition phenomena considered in a MACCS analysis.
(Taken from NUREG-6244)

To capture the stochastic uncertainty, the results of analyses are often plotted on a curve called a complementary cumulative distribution function (CCDF). A CCDF curve plots the probability of exceeding a specific consequence (such as prompt fatalities or latent cancers per reactor year) as a function of that consequence (number of prompt fatalities or latent cancers) using 1 years' worth of weather data. This is simply an alternative form of displaying the information in Fig. 2.

The second type of uncertainty is called "state of knowledge uncertainty." This encompasses aspects such as the model uncertainty. This includes the completeness of identifying and modeling all significant phenomena, using the appropriate inputs on each phenomenon. There may be a range of potentially valid input parameters for various phenomena, and each analyst may have different assumptions and values that they think are most appropriate. Depending on which assumptions are used, and which input values are selected, the uncertainty in the calculations can change and the output results may vary widely. To illustrate this point, the MACCS (and MACCS2) model which is widely used in Level 3 PRAs uses a Gaussian plume model. According to *MACCS2 Guidance Report* (DOE-EH) (U.S. DOE, June 2004), that model was estimated to be within a factor of 2 for distances from 100 m to 10–20 km and to be accurate to about an order of magnitude (factor of 10) for distances beyond that. Certain atmospheric conditions can introduce even greater uncertainty. Additionally, there are other MACCS input parameters that have been shown to impact the consequence analyses, such as the scaling factors for horizontal and vertical radionuclide dispersion, the dry deposition velocity, and the groundshine shielding factor. Another factor impacting consequence analysis would be the fact that the downwind population varies throughout the day, and consequences depend heavily upon the impacted population and population density.

Thus, understanding the accident modeling uncertainties is very important when trying to understand the potential impacts of any accident or the changes that certain parameters (LEU vs MOX) may have on the consequences of an accident. For example, if a source term varied within 10–20% of an estimated value, this factor would not be significant when compared to the uncertainties inherent in the weather modeling, population density estimates, population exposure times, and a number of other important assumptions that go into a consequence analysis.

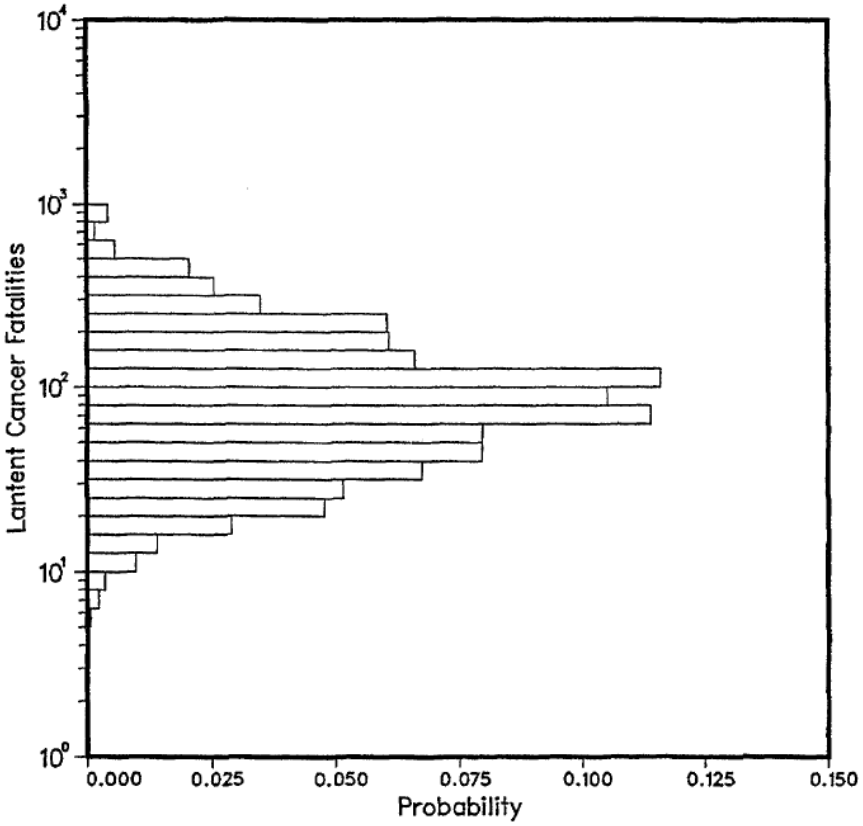

Fig. 2. Distribution of latent cancer fatalities for a specific source term (STG SUR-49).
(Taken from the NUREG-1150 Main report, Figure B.16)

4. STORAGE HANDLING AND TRANSPORTATION

The isotopic composition, decay heat, gamma and neutron radiation fields, and toxicity are dependent on specific details and parameters and characteristics of particular fuel cycles and burnup levels, but in general, there is little difference between spent LEU fuel and spent MOX fuel for these parameters. If a MOX fuel assembly decay heat output is not bounded by a corresponding UO_2 assembly of similar burn up, age out of reactor, additional analyses and operational constraints may be imposed to demonstrate that MOX assemblies meet regulatory limits and criteria for the dry storage system being used. For example, it may be operationally necessary to keep MOX fuel assemblies in spent fuel pools longer (aging/cooling) prior to transfer to an Independent Spent Fuel Storage Installation (ISFSI) in order to ensure maximum fuel cladding temperature limits are met for normal conditions of storage and short-term loading operations. (IAEA, 2011; Ade and Gauld, 2011)

5. SUMMARY

Although this paper provides a short history and discussion of some of the important issues pertaining to WG-MOX, the primary focus of the paper is to provide a useful bibliography of resources pertaining to weapons-grade MOX fuel use in LWRs. Therefore, not all aspects of MOX use were reviewed in this paper.

Sandia National Laboratory and other investigators have performed significant work in the United States on severe accident source terms for MOX fuel cycles in comparison with LEU fuel cycles. Sandia's source term reports indicate that due to some off-setting factors, the actual source terms between MOX and LEU are not significantly different despite the inventory differences. However, those source terms were not used to perform an explicit consequence evaluation to draw a conclusion on the overall integrated risk. Even though there are inventory differences and there is the potential for differences in the source term for MOX fuel, the overall uncertainties in the accident progression, fission product release/retention, source term and risk analyses would indicate that there is no significant difference between MOX and LEU fuel. In addition, recent research shows that current guidelines for fission product release (NUREG-1465) are significantly conservative (by a factor of 2).

The National Academy of Sciences has stated:

> "We believe, further, that under these circumstances no important overall adverse impact of MOX use on the accident probabilities of the LWRs [light water reactors] involved will occur, if there are adequate reactivity and thermal margins in the fuel, as licensing review should ensure, the main remaining determinants of accident probabilities will involve facts not related to fuel composition and hence unaffected by the use of MOX rather than LEU Fuel (NAS, 1995)."

ANNOTATED BIBLIOGRAPHY OF SELECTED MOX DOCUMENTS

Summary reviews are presented here for **selected** relevant MOX documents, using the bibliography in Appendix A.

Ade, B. J., and I. Gauld, *Decay Heat Calculations for PWR and BWR Assemblies Fueled with Uranium and Plutonium Mixed Oxide Fuel Using SCALE*, ORNL/TM-2011/290, Oak Ridge National Laboratory, Oak Ridge, TN, September 2011.

Decay Heat Calculations for PWR and BWR Assemblies Fueled with Uranium and Plutonium Mixed Oxide Fuel Using SCALE presents decay heat rates for UOX, reactor-grade MOX, and weapons-grade MOX for a typical PWR and a typical BWR fuel assembly. ORIGEN-ARP libraries were generated for both the PWR and BWR using a specific and identical fuel composition. In order to obtain the approximate equivalent PuO_2 content in MOX compared to a given ^{235}U enrichment in UOX, an interpolation procedure was performed. The approximate equivalent PuO_2 content was used in SCALE/ORIGEN-ARP calculations to calculate decay heat rates up to nearly 20,000 days after discharge from the reactor. The PWR and BWR calculated decay heat rates were similar. In general, using MOX fuel results in a higher specific decay heat rate than UOX. Reactor-grade plutonium in MOX fuel results in higher decay heat than using weapons-grade plutonium in MOX fuel. The crossover points for the PWR fuel assemblies at all discharge burnup points analyzed are less than 0.25 days for RG-MOX, and less than 0.9 days for WG-MOX. For the BWR fuel assembly, the crossover points at all discharge burnup points analyzed are less than 0.75 days for RG-MOX, and ~2.0 days for WG-MOX. The decay heat ratios at decay times greater than 10 days are very similar for the PWR and BWR fuel bundles. In general, the decay heat ratio (MOX/UOX) is proportional to the decay time after discharge and inversely proportional to the discharge burnup; the decay heat ratio increases with increasing time after discharge and decreases with increasing discharge burnup.

Akkurt, H. and N. M. Abdurrahman, *Neutronics Benchmarks for the Utilization of Mixed-Oxide Fuel: Joint U.S./Russian Progress Report for Fiscal Year 1997; Volume 4, Part 5—ESADA Plutonium Program Critical Experiments: Multiregion Core Configurations*, ORNL/SUB/00-XSZ175V-1, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, Oak Ridge, TN, 2000.

ESADA Plutonium Program Critical Experiments: Multiregion Core Configurations details a set of critical experiments done using MOX and UO_2 fuels. For the program, 88 unique experimental configurations were setup. The two MOX fuels contained 8 wt % ^{240}Pu and 24 wt % ^{240}Pu, respectively, and the UO_2 fuel had 2.72 wt % enrichment. The critical experiment results were compared to MCNP calculations using both ENDF/B-V and ENDF/B-VI data. Generally, the MCNP calculations resulted in k_{eff} values within 0.5% of the critical experiments' values.

Akkurt, H. et al., *Neutronics Benchmarks for the Utilization of Mixed-Oxide Fuel: Joint U.S./Russian Progress Report for Fiscal Year 1997; Volume 4, Part 6—ESADA Plutonium Program Critical Experiments: Power Distribution Measurements*, ORNL/SUB/99-XSZ175V-1, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, Oak Ridge, TN, 2001.

Volume 4, Part 6 of the *Neutronic Benchmarks for the Utilization of Mixed-Oxide Fuel: Joint U.S./Russian Progress Report for Fiscal Year 1997* studied the ESADA Plutonium Program Critical

Experiments: Power Distribution Measurements. It summarizes the critical experiments that were done at the Westinghouse Reactor Evaluation Center (WREC) in 1967 using mixed-oxide (MOX) PuO_2-UO_2 and/or UO_2 fuels in various lattices and configurations.

Two variations of MOX fuel and a low-enriched UO_2 fuel were the subjects of the critical experiments. The two MOX fuel variants had a wide range in ^{240}Pu content: 8 wt % and 24 wt %, respectively. The critical experiments in the ESADA program included 88 unique critical core configurations that involved both single and multiregion core configurations. The configurations were modified by varying the lattice pitch, the fuel configuration, and the fuel isotopic composition. The relative error between experimental and calculated MCNP power values was generally below 2%.

Anderson, C. K. and R. H. Klinetob. "Plutonium Burning Light Water Reactor Concept," presented at the ANS/CAN Joint Meeting, 1976.

"Plutonium Burning Light Water Reactor Concept" looked at the possibility of using Combustion Engineering's nuclear reactor as a plutonium burner. CE's reactor was capable of operating with a core completely full of MOX fuel. Furthermore, it was found that in the all-plutonium mode, the reactor could utilize recycled plutonium from two PWRs in addition to the plutonium from earlier cycles of the reactor.

It describes the Combustion Engineering System-80 Reactor System and provides a quantitative comparison of standard uranium–oxide fuel characteristics with those for self-generated recycle (~1/3-core MOX), and also full-core MOX fueling. The focus in the report is on the unique control rod design which essentially makes all core positions accessible, and, therefore, removes the concerns for its control and operation associated with loss of reactivity worth with MOX fuel. As a result, the paper concludes that "the consequences of anticipated operational occurrences and other postulated incidents are no more severe in the plutonium burner than in the uranium-fueled or self-generated plutonium recycle modes of operation, provided that the reactivity control systems are designed to compensate for the reduction in control poison effectiveness."

The study found that the benefits of a plutonium burner included lower plutonium inventories, reduced safeguards concerns because of more centralized concentration, helping reprocessing development, and reducing the demand for other energy fuels.

Aniel-Buchheit, S., et al., "Plutonium Recycling in a Full-MOX 900-MW(electric) PWR: Physical Analysis of Accident Behaviors," *Nuclear Technology*, **128**, pp. 245–256 (1999).

"Plutonium Recycling in a Full-MOX 900-MW(electric) PWR: Physical Analysis of Accident Behaviors" analyzes an all-MOX-fueled reactor core PWR. One main goal of the report was to compare the neutronic specificity and characteristics of plutonium nuclear fuel with those of uranium fuel.

This paper discusses the feasibility of 100% MOX nuclear fuel recycling into a PWR. The report contains information on the advantages of full-MOX core loading, neutronics features, and reactivity coefficients compared to a standard UO_2-fueled reactor core and behavior during accident scenarios (particularly regarding the effects of moderator density). Through studying accident scenarios with a change in moderator density, it was concluded that plutonium fuel can be used in 900-MW(e) PWR nuclear power plants as long as the fuel is not of very poor quality (i.e., high in ^{240}Pu).

Ashbaugh, S. G., et al., *Simulation of Mixed Oxide (MOX) Versus Low Enrichment Uranium (LEU) Fuel Severe Accident Response Using MELCOR*, SAND 2005-4361c, Sandia National Laboratories, Albuquerque, NM, 2005.

Simulation of Mixed Oxide (MOX) Versus Low Enrichment Uranium (LEU) Fuel Severe Accident Response Using MELCOR documents calculations for LEU and 40% MOX cores for accident categories identified as frequency- or risk-dominant. The results indicated that under expected severe accident conditions, volatile fission product releases occur at very high release rates owing to high fuel temperatures, regardless of the fuel type. Hence, differences noted in experimental results at lower temperatures do not produce appreciable differences in net volatile releases under severe accident conditions.

From the summary and conclusions section of the paper: "For most accident scenarios and fission product types, LEU release rates were slightly higher than 40% MOX cases for two reasons. First, the LEU core had a higher average and maximum assembly burnup, increasing the core-wide fission product inventory. Consequently, for the same fission product release fraction, LEU mass released is higher. Second, higher LEU burnup fuel had a higher decay power. Therefore, the accident progressed somewhat more quickly with somewhat higher fuel temperatures (i.e., leads to higher fission product releases). Observed differences in release quantities are considered to be less than variations due to more global accident progression uncertainties.

In some specific cases and fission product classes, 40% MOX release rates were higher than LEU release rates. These differences can be traced to two causes. For some fission products, the 40% MOX inventory was higher than the LEU inventory (e.g., the "tetravalent" fission product class, including plutonium). Second, global phenomenological uncertainties caused variations in calculated accident progression contributing to different localized release rates for some sequences. In particular, some late phase uncertainties can cause significant changes in the accident response and late fission product releases. The series of calculations represent point estimates to the calculated response. However, previous studies which systematically evaluated severe accident uncertainties (Gauntt et al., 2002) show variations in LEU response that bounds the differences noted in the present report. Hence, no significant fission product release differences were noted in the LEU and 40% MOX severe accident response for the wide range of risk-significant transients considered.

Conclusions do not evaluate whether there is an increased risk in operation of specific nuclear plants with a 40% MOX core. Differences in specific activities, as well as other issues, would need to be considered in a comprehensive risk evaluation. Comparison of source terms calculated for LEU cores against 40% MOX cores indicates little difference in terms of released mass for any given MELCOR release class.

Ashbaugh, S. G., et al., *Assessment of Severe Accident Source Terms in Pressurized-Water Reactors with a 40% Mixed-Oxide and 60% Low-Enriched Uranium Core Using MELCOR 1.8.5*, SAND 2008-6665, Sandia National Laboratories, Albuquerque, NM, 2010.

"Assessment of Severe Accident Source Terms in Pressurized-Water Reactors with a 40% Mixed-Oxide and 60% Low-Enriched Uranium Core Using MELCOR 1.8.5" documents a series of 23 severe accident calculations using MELCOR for a four-loop Westinghouse reactor with an ice condenser containment. The calculations covered five accident classes identified as the risk- and consequence-dominant accident sequences in plant-specific PRAs for the McGuire and Catawba nuclear plants. These were:

1. long-term SBO with late containment failure,
2. long-term SBO with early containment failure,
3. SLOCA with failure to realign ECCS for recirculation and late containment failure,
4. SLOCA with failure to realign ECCS for recirculation and early containment failure, and
5. LLOCA with failure of ECCS and late containment failure.

Calculations were performed for traditional LEU core loading and a core loaded with 40% MOX fuel assemblies and 60% LEU fuel assemblies. The End of Cycle (EOC) fission product inventory and decay

heat distribution for the 40% MOX core were determined based on planned administrative limits as described in the Pu disposition plans for the nuclear plants.

In general, the accident progression and source terms for the LEU and 40% MOX cases were similar. The MELCOR calculations show that at severe accident fuel temperatures, the volatile fission product releases occur at a very high release rate, regardless of the fuel type. The differences noted in the experimental results at lower temperature were not prototypical of severe accident conditions in the long term and did not greatly impact the integral source term.

Plant-specific PRAs were used to identify important uncertainties in accident progressions in the five accident categories described above. Sensitivities examined included failure of AFW, coincident RCP seal failure with SBO, pressurizer SORV, LOCA break size, and LOCA break location. For each calculation performed, source term timing results are presented in a form based on the NUREG-1465 prescription. That is, timing information is presented in terms of:

- onset of release,
- duration of coolant activity release,
- duration of gap release,
- duration of in-vessel release,
- duration of ex-vessel release, and
- duration of late in-vessel release.

The collection of the source term results for all of the 40% MOX core MELCOR calculations can be used as the basis to develop a representative source term (across all accident types) that will be the MOX supplement to NUREG-1465.

Ashbaugh, S. G., et al., *Accident Source Terms for Pressurized Water Reactors with High-Burnup Cores Calculated Using MELCOR 1.8.5*, SAND 2008-6664, Sandia National Laboratories, Albuquerque, NM, 2010.

Accident Source Terms for Pressurized Water Reactors with High Burnup Cores Calculated Using MELCOR 1.8.5 documents the investigation as to whether the NUREG-1465 source term for PWRs requires an update. MELCOR 1.8.5 is used to model fission gas releases following hypothetical severe accident scenarios in a PWR. The report concluded that more fission products are retained by the vessel than was reported in NUREG-1465, and, thus, fewer fission products are predicted to be released to the containment. Overall, it was found that NUREG-1465 overestimated the release to containment by a factor of 2.

Bairiot, H. and C. Vandenberg, "Use of MOX Fuels: The Reasons to Start," pp. 65–95 in *Nuclear Fuel Cycle in the 1990s and Beyond the Century—Some Trends and Foreseeable Problems*, Technical Reports Series 305, International Atomic Energy Agency, Vienna, Austria, 1989.

"Use of MOX Fuels: The Reasons to Start" looks at the technical and economic reasons that suggest and support the utilization of MOX nuclear fuel in thermal reactors. The report concluded that MOX fuel behaves as well as uranium fuel. It further concluded that international regulations need to allow used fuel recycling at the required rate because of the use of MOX fuel. The report also recommends that storing plutonium is expensive and should be kept to a minimum for both economic and safeguards reasons.

This report provides technology data for plutonium and MOX fuel utilization in thermal nuclear reactors. It is noted in the report that most of the radiological hazard associated with the MOX fuel disappears once the material is in the fuel rods. There are storage problems arising from the decay heat generated by alpha activity (e.g., from ^{238}Pu). Radiation shielding designs should account for and consider the neutron

activity (Chapter 3). It also discusses neutronic characteristics; thermal, mechanical, and chemical properties; fabrication processes; design criteria; and the safety study.

Bay, H. and R. Stratton, "Use of Mixed Oxide Fuel in a Pressurized Water Reactor: Experience of NOK, Switzerland," pp. 292–297 in *Proceedings of the International Topical Meeting on Safety of Operating Reactors*, 1998.

"Use of Mixed Oxide Fuel in a Pressurized Water Reactor: Experience of NOK, Switzerland" describes the experience of NOK with using MOX fuel in two PWRs in Beznau, Switzerland. They have determined and instituted design criteria such that the MOX assemblies have the same safety, reliability, and energy production capabilities of uranium fuel assemblies. NOK allowed and is licensed for the use of up to 40% MOX core loadings at the time the paper was released (1998). The paper also discusses the future of plutonium recycling and MOX fuel fabrication.

The safety of Operating Reactors in the 1998 timeframe is discussed in the context of the license that allows up to 40% MOX fuel core loading. There is a section in the report on "experience with Transport, Handling, and Safeguards." The paper also discusses the inspection process and strategies at the nuclear power plant. With the aging of MOX fuel, some minimal radiation shielding is required to reduce the personnel exposure for those in close proximity to the fuel. Otherwise, no extra precautions are taken above what prudent radiation protection practices require. The report points out that no special steps or procedures are required for the storage of spent MOX fuel compared to the usual LEU UO_2.

Belov, A., et al., *Calculation Analysis of San Onofre Depletion MOX Fuel Experiment*, ORNL/SUB/00-85B99398V-8, Russian Research Center "Kurchatov Institute" Institute of Nuclear Reactors, Oak Ridge National Laboratory, Oak Ridge, TN, 2001.

Calculation Analysis of San Onofre Depletion MOX Fuel Experiment compares the calculational results for the isotopic compositions in the San Onofre reactor core fuel with measured values. MOX fuel that was irradiated at the San Onofre PWR nuclear power plant was the source of the measured data, and the Kurchatov Institute's MCU/BURNUP Monte Carlo code was used in generating the calculational results data. Analysis of the results found that the predicted and measured plutonium content did not differ by more than 3%. The main source of error was related to the ^{239}Pu isotope.

Besnainou, J., AREVA Inc., letter to Energy Reporters, April 15, 2011.

"Open Letter to Energy Reporters: Stop the Sensationalism on MOX Fuel!" addresses the issues and agenda in media reporting on MOX fuel usage in nuclear reactors after the accident situation at Fukushima Dai-ichi Unit #1 to #4 in Japan following the March 2011 earthquake and tsunamis. This letter expressed the opinions of the CEO of AREVA that the news coverage of MOX fuel has been plagued with inaccuracies and half-truths. The CEO of AREVA (Jacques Besnainou) points out that the situation in Japan was not caused by MOX fuel and has not caused the situation to be worse than it would have been otherwise. The letter points out that MOX fuel is safe, has been tested, and is cost effective. It also discusses the nonproliferation benefits of using MOX fuel.

Brown, O. C., et al., "Safety Analysis for Mixed Oxide (MOX) Fuel in Boiling Water Reactors (BWRs)," pp. 282–287 in *Proceedings of the International Topical Meeting on Safety of Operating Reactors*, 1998.

"Safety Analysis for Mixed Oxide (MOX) Fuel in Boiling Water Reactors (BWRs)" investigates the MOX nuclear fuel Atrium-10 assembly design. The study concluded that this fuel design is acceptable for both 12- and 24-month BWR cycles.

The main part of the paper deals with BWRs. The report states that the mechanical structures of the UO_2 and MOX fuel assemblies are the same. Regarding safety-related reactor systems, the paper indicated that there is no need for a system modification due to loading of MOX fuel.

The 12- and 24-month BWR cycles were shown to be acceptable through analyses, and the 12-month fuel cycle was shown to be acceptable through the operating experience. The memo report also discusses the analysis codes developed by Siemens to analyze the MOX fuel (i.e., CASMO-3G/MICROBURN-B2).

Cowell, B. S., and S. E. Fisher, *Survey of Worldwide Light Water Reactor Experience with Mixed Uranium-Plutonium Oxide Fuel,* ORNL/TM-13428. Oak Ridge National Laboratory, Oak Ridge, TN, 1999.

Survey of Worldwide Light Water Reactor Experience with Mixed Uranium-Plutonium Oxide Fuel reviews government- and industry-sponsored programs and experience in the United States, Belgium, Italy, Germany, France, Switzerland, Japan, and the United Kingdom from the 1960s through the 1990s. It provides an overview of existing MOX fuel manufacturing steps. The plutonium used in much of the domestic research and development had a high fissile content, similar to that of WG plutonium. The French, Belgian, and German MOX programs led to the development and commercial utilization of state-of-the-art MOX fuels by the mid-1980s.

Debes, M., "Nuclear Fuel Management Policy in EDF PWRS: MOX Achievements and Safety Related Issues," pp. 288–291 in *Proceedings of the International Topical Meeting on Safety of Operating Reactors*, 1998.

"Nuclear Fuel Management Policy in EDF PWRS: MOX Achievements and Safety Related Issues" details the nuclear fuel management policy in France in EDF PWRs. In particular, it looks at advancements in MOX nuclear fuel management. It further describes the studies related to accident scenarios and safety concerns and issues in regards to MOX nuclear fuel. The paper expresses the hope that further studies relating to MOX will show what they call "MOX parity" in fuel management and burn-up behavior.

The report focuses on the safety of operating reactors around 1998. Discussions in the report include the achievements gained in MOX fuel development and management.

Desmoulins, P., "EDF MOX Fuel Operating Experience," presented at the *27th International Utility Nuclear Fuel Performance Conference*, Idaho Falls, ID, August 5–8, 1996.

"EDF MOX Fuel Operating Experience" describes the experience gained in France by EDF through the use of MOX fuel in their nuclear reactor fleet (up to 1996). Their experience concluded that various factors of reactor core behavior using MOX fuel are equivalent to that of UO_2 fuel. The reliability of MOX fuel is shown to be acceptable by the operating experience that was gained. The operating experience includes 30 completed or in-progress irradiation cycles which took place in over seven different nuclear reactors.

The report contains brief discussions on fuel handling, examination of fresh fuel, and reactor vessel neutron fluence levels. The report noted that MOX core behavior is equivalent to that of UO_2 cores in terms of various operating and safety parameters.

Ellis, R. J., *System Definition Document: Reactor Data Necessary for Modeling Plutonium Disposition in Catawba Nuclear Station Units 1 and 2*, ORNL/TM-1999/255, Oak Ridge National Laboratory, Oak Ridge, TN, 2000.

System Definition Document: Reactor Data Necessary for Modeling Plutonium Disposition in Catawba Nuclear Station Units 1 and 2 outlines the information needed to model the Westinghouse-type PWR nuclear reactors at Catawba Units 1 and 2. In particular, it includes data about the reactor operating details and conditions, fuel assembly data, IFBA pellet data, burnable poison rod (BPR) data, control rod data, and reactor core baffle data. The fuel assemblies are of the 17 × 17 fuel rod design. The report also lists information about the MOX fuel assemblies including fuel pin plutonium content and fuel management strategies as considered in the 1999/2000 time frame. The work documented in this report was part of the project that would lead to the irradiation of MOX Lead Test Assemblies (LTA) at Catawba.

Ellis, R. J., et al., *Support to TVA Data Needs for MOX Core Designs*, ORNL/GNEP/LTR-2008-077, Oak Ridge National Laboratory, Oak Ridge, TN, 2008.

The DOE-NE sponsored report *Support to TVA Data Needs for MOX Core Designs (GNEP)* assesses the modifications necessary to use MOX nuclear fuel in a four-loop Westinghouse PWR. Information is presented on reactor core physics, neutronics, source terms, fuel isotopic, and fuel handling.

The report includes sections about UO_2 and MOX models, SNF and reactor-grade plutonium, radiological source terms, decay heat, and physics data about MOX core transient analysis. The report also includes information about equipment lifetime impacts, fuel design parameters, core design impacts, fuel handling and storage impacts, new fuel/irradiated fuel shipment impacts, fuel utilization impacts and cost, and fuel storage impacts.

Emmett, M. B., *Calculational Benchmark Problems for VVER-1000 Mixed Oxide Fuel Cycle*, ORNL/TM-1999/207, Oak Ridge National Laboratory, Oak Ridge, TN., 2000.

Calculational Benchmark Problems for VVER-1000 Mixed Oxide Fuel Cycle compares Russian and American computational methods and data. In particular, it deals with the storage and handling of VVER nuclear reactor fuel. The following quantities and neutronics characteristics were analyzed and assessed about fresh fuel of low-enriched uranium fuel, weapons-grade MOX fuel, and reactor-grade MOX fuel—multiplication factor, neutron source, and neutron dose rates. The results indicated that differences between the fuel types were less than 2% for multiplication factors, less than 1% for the neutron source rate, and less than 9% for the neutron dose rate.

Fisher, S. B., et al., "Microstructure of Irradiated SBR MOX Fuel and Its Relationship to Fission Gas Release," *Journal of Nuclear Materials*, 306, pp. 153–172 (2002).

The journal article "Microstructure of Irradiated SBR MOX Fuel and Its Relationship to Fission Gas Release" investigated the microstructure and microchemistry of SBR MOX nuclear fuel compared to UO_2 fuel. The investigation concluded that the factors that influence gas release (grain size, porosity, and dislocation density) do not have a perturbation outside of the general boundary or the variability is natural

in intergranular bubble formation. The study also found that nearly identical fission gas release occurs when the SBR MOX fuel was compared to traditional UO_2 fuel.

Frankel, A. J., et al., "PWR Plutonium Burners for Nuclear Energy Centers," presented at the ANS Topical Meeting, 1981.

"PWR Plutonium Burners for Nuclear Energy Centers" (NEC) studies the technical issues related to Nuclear Energy Centers. The proposed Nuclear Energy Center that was studied included a nuclear reactor on the same site as a reprocessing facility and a mixed-oxide nuclear fuel fabrication facility. This eliminates the need to ship plutonium off-site for fuel preparation.

This report is essentially an expanded description of the Combustion Engineering System-80 reactor system. In addition to addressing most of the items discussed in the body of this report, the paper also identifies six "considerations which give rise to design modifications," including increased heating in the nuclear reactor due to higher gamma source strength, and a reduction in the effective lifetime of fixed neutron sources, or a need for additional or stronger neutron sources. The focus in this report is on the full-core implementation of MOX, and the cycle lengths are relatively low compared to current practice.

The paper also studies the effects of increased plutonium loadings on the nuclear reactor design and the safety analysis of the reactor. Combustion Engineering's Nuclear Steam Supply System was also investigated. The ability to have different amounts of plutonium was seen as an advantage to the NEC reactor.

Gauntt, R. O., et al.. *Synthesis of VERCORS and Phebus Data in Severe Accident Codes and Applications*, SAND 2010-1633, Sandia National Laboratories, Albuquerque, NM, 2010.

Synthesis of VERCOR and Phebus Data in Severe Accident Codes and Applications updates the original default specifications given for the MELCOR code. Separate instructions are given for the modeling of in-core and spent fuel pool releases. The user is given inputs that can manually override the current MELCOR defaults until the new parameters can be incorporated into that older version of MELCOR. Some of the deficiencies that were raised by a variety of other reports are addressed by the new approach given in this report.

Gauntt, R. O., *Phenomenological Advances of Severe Accident Progression*, prepared for the U.S. Nuclear Regulatory Commission, Sandia National Laboratories, Albuquerque, NM, 2009.

The power point presentation *Phenomenological Advances of Severe Accident Progression* looks at the work done with NRC's State-of-the-Art Reactor Consequence Analysis. Its objective was to "perform a state-of-the-art, realistic evaluation of severe accident progression, radiological releases and offsite consequences for important low likelihood accident sequences." The models developed account for both the improvements in modern nuclear power plant design and significant operator actions. Some of the conclusions that were obtained through using the improved code included significantly smaller source terms, more time possible for mitigating the accident, more fission products retained in the RCS and containment, and slower-developing accidents.

Gehin, J. C., J. J. Carbajo, and R. J. Ellis, *Issues in the Use of Weapons-Grade MOX Fuel in VVER-1000 Nuclear Reactors: Comparison of UO_2 and MOX Fuels*, ORNL/TM-2004/223, Oak Ridge National Laboratory, Oak Ridge, TN, 2004.

In this report the significant differences between MOX and UO_2 fuel as used in VVER-1000 reactors were identified and discussed. A summary of the key issues follows referencing the different chapters of the report:

Assembly and Core Designs (Chap. 2)

- Suitable assembly and core designs can be obtained for the current 12-month fuel cycle as well as for the 18-month increased disposition rate cores.
- The validation data available for certification of design codes is sufficient for insertion of LTAs.
- Additional data to validate data and codes for absorber reactivity worth, reactivity coefficients, and isotopic composition are needed.

Neutronics Properties (Chap. 3)

- Plutonium-239 has significantly different nuclear properties than ^{235}U that impact the neutronic behavior of the reactor core.
- The assembly reactivity vs burnup is different for MOX fuel than UO_2 fuel; but with proper design, this difference does not affect the operation of the core in terms of fuel cycle length.
- The neutron spectrum resulting from MOX fuel is harder than that from UO_2 fuel. This harder spectrum reduces the worth of the soluble boron and control rod absorbers.
- The worth of absorber materials (boron and gadolinium) used in the soluble boron, burnable absorbers, and control rods is lower in MOX cores. This results in the increase in the soluble boron concentration, increased use of burnable absorbers, and a modification to the control rods to use enriched boron.
- The power distribution and linear heating rates in MOX cores are comparable to that of UO_2 cores.
- The delayed neutron fraction and prompt neutron lifetime are smaller in MOX cores and will require analysis to determine any impacts on the reactor safety analysis.
- The moderator temperature coefficient of reactivity is more negative in MOX cores, particularly at BOC than UO_2 cores. The Doppler coefficient of reactivity is also slightly more negative.
- The fast flux at the pressure vessel is not significantly increased in MOX cores if a low-leakage loading pattern with UO_2 assemblies on the core periphery is used.

MOX and UO_2 Thermophysical Properties (Chap. 4)

- The difference in thermal conductivity (lower in MOX fuel) results in a higher MOX fuel temperature (about 50–100 K centerline) and energy stored as compared to UO_2 fuel at the same power levels.
- This higher temperature increases the release of fission products into the gap of MOX fuel.
- MOX fuel has more inhomogeneities, lower heat of fusion, and lower melting temperature than UO_2 fuel.

Decay Heat (Chap. 5)

- Decay heat from MOX fuel is lower during the first day or days after irradiation and larger after that than UO_2 fuel.
- The lower decay heat of MOX fuel in the short term is beneficial when compared to UO_2 fuel for accidents like LOCAs.
- MOX fuel decay heat is larger than UO_2 decay heat in the long term (days), and this may be detrimental in severe accidents and in cooling spent fuel when compared to UO_2 fuel.

Source Terms (Chap. 6)

- Irradiated MOX and UO_2 fuels have comparable total fission product activities with variations being nuclide dependent based on the differing fission production yields. MOX source term has more actinides and more iodines, which may result in higher neutron and gamma doses from MOX fuel. Gap source terms may be larger for MOX fuel, depending on operating temperatures.
- Irradiated MOX fuel has a significantly larger inventory of actinides, which will be significant at long decay times when actinides dominate the source term.
- Severe accidents consequences (and doses) may be different and possibly worse (higher) for MOX fuel.

Probability Risk Assessment (Chap. 7)

- PRA levels 1, 2, and 3 need to be revised/upgraded for cores with MOX fuel.

Accident Analysis (Chap. 8)

- The results of the design basis accidents completed have shown that MOX fuel can be safely burned in VVER-1000 reactors without significant differences compared to UO_2 fuel. Cores with up to 1/3 MOX were used in these calculations and only the control rods were upgraded.
- Other accidents and calculations for cores with 41% MOX need to be completed to investigate the need for other reactor modifications. Severe accidents need to be evaluated also to assess the difference in source term releases and doses when compared to UO_2 fuel.

Fresh Fuel Criticality and Dose Properties (Chap. 9)

- There is no significant difference in the criticality of MOX and UO_2 fuel in transportation, handling and storage.
- Weapons-grade MOX has significantly higher gamma and neutron dose rates than UO_2 fuel. These higher dose rates must be considered in the suitability of existing transportation package sets, which may require modifications. The higher dose rates must also be considered in handling and storage but are sufficiently low that modifications to the plant may not be required.
- MOX has a significantly higher alpha activity than UO_2 fuel. The cladding is sufficient to contain this activity and protect the fuel handlers for intact and undamaged fuel. However, additional monitoring and storage will be required for damaged fuel.

Spent Fuel Criticality and Dose Properties (Chap. 10)

- The criticality of spent MOX fuel is generally the same as for spent UO_2 fuel and is bounded in safety calculations by the assumption of fresh, unburned fuel.
- Spent MOX fuel has a significantly larger neutron source and a comparable gamma source in comparison to spent UO_2 fuel.
- For the spent fuel cask, the higher neutron source results in a factor of 3 increase in the shipping cask external dose rate. Therefore, additional neutron shielding will be required to maintain the same dose rate as for UO_2 fuel.
- Alternatively, mixed loadings of the TK-13 cask should be considered in which the interior assemblies are MOX, and the exterior assemblies are UO_2. This configuration would provide the additional shielding required.

Security and MPC&A (Chap. 11)

- Because MOX fuel contains WG plutonium that can be diverted to weapons use, additional security and control accountability measures must implemented with MOX fuel.

Gomin, E., et al., *VENUS-2 Experimental Benchmark Analysis with MCU-REA*, ORNL/SUB/00-85B99398V-5, Russian Research Center "Kurchatov Institute" Institute of Nuclear Reactors, Oak Ridge National Laboratory, Oak Ridge, TN, 2001.

VENUS-2 Experimental Benchmark Analysis with MCU-REA documents the results of the analysis using the Russian code MCU-REA. The VENUS-2 experiment involved the use of MOX fuel in the VENUS critical facility zero power reactor at SCK-CEN in Belgium. Various different fuel compositions were used including 3.3 wt % UO_2, 4.0 wt % UO_2, and 2.0/2.7 wt % MOX. The report shows the comparison of results of the MCU-REA code (continuous energy Monte Carlo code system) with the results of other codes and with experimental data. The report concluded that the MCU-REA code produced results consistent with comparable codes and with the experimental data. The exact deviation of the calculated k_{eff} value for the VENUS-2 critical facility was –0.35%.

Hermann, O. W., *Benchmark of SCALE (SAS2H) Isotopic Predictions of Depletion Analyses for San Onofre PWR MOX Fuel*, ORNL/TM-1999/326, Oak Ridge National Laboratory, Oak Ridge, TN, 2000.

Benchmark of SCALE (SAS2H) Isotopic Predictions of Depletion Analyses for San Onofre PWR MOX Fuel benchmarks the Scale/SAS2H code sequence (in an older version of Scale) using isotopic analyses of the MOX spent fuel taken from the San Onofre PWR, Unit 1. Using either mass or alpha spectroscopy, isotopic analyses were performed on 13 actinides and ^{128}Nd in the MOX spent fuel.

The data for the analyses were taken from six fuel pellet samples that were in four different fuel pins. When the code results and isotopic analyses were compared, the resulting differences were –0.9% for ^{235}U and 5.2% for ^{239}Pu. Overall, the mass spectrometer analyses produced fewer differences than the alpha spectrometer analyses. The differences seen in the isotopic determinations between the measured and calculated values were generally larger in the MOX-fueled than in uranium-only fueled reactors. In comparison to measured values, the average percentage differences of the calculated values were –0.9% for ^{235}U and 5.2% for ^{239}Pu.

IAEA, *Status and Advances in MOX Fuel Technology*, Technical Reports Series No. 415, International Atomic Energy Agency, Vienna, Austria, 2003.

Status and Advances in MOX Fuel Technology provides an overview of the worldwide state of plutonium fuel development as of December 2000. Information on the, MOX fuel technology in the areas of plutonium feed production, handling, and storage; LWR fuel assembly design; fuel fabrication; performance; in-core fuel management including reload strategies; MOX impact on normal reactor operations; transportation; spent MOX fuel management; decommissioning, waste treatment, safeguards and alternative approaches for plutonium recycling is provided. The report concentrates on MOX fuel for thermal power reactors.

IAEA, *Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management*, IAEA Nuclear Energy Series No. NF-T-3.8, International Atomic Energy Agency, Vienna, Austria, 2011.

Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management reviews the fuel characteristics of UOX and MOX; the analysis is focused on the back end of the fuel cycle, comparing lower burnup UOX fuel and cladding types from LWRs and HWRs having zirconium alloy based cladding and structural materials to either higher burnup UOX or MOX for the same reactors and cladding types. It notes that the mechanical designs of lower burnup UOX and higher burnup UOX or MOX fuel are very similar, but some of the properties of higher burnup UOX and MOX are potentially different. Examples of differences in properties between lower burnup UOX and higher burnup UOX and MOX include: higher fuel rod internal pressures; higher decay heat; higher specific activity; and degraded cladding mechanical properties. Evaluation of these effects on the back end of the fuel cycle was based on regulatory and safety criteria, and operational and economic considerations. Higher burnup UOX or MOX usage affects all spent fuel management components, such as wet and dry storage, transportation, reprocessing, re-fabricated fuel, and final disposal.

The evaluation spans a wide range of issues and factors that need to be considered with respect to the use of higher burnup UOX or MOX were identified. Analysis of the different spent fuel management components revealed:

— **Wet storage:** Because wet storage is associated with low temperatures, the clad material property degradation is expected to be low. High burnup UOX and MOX storage will increase the heat load and potentially radioactive releases. This may require an upgrade of the pool facility with respect to heat removal, pool cleanup systems, and additional neutron poison material in the pool water or in storage racks. Re-evaluation of criticality and regulatory aspects may also be required.

— **Dry storage:** In dry storage, the cask has to provide safe confinement/containment and, in parallel, the decay heat has to be removed to limit temperature-induced material alterations. This means, dry storage is more sensitive to increased UOX burnup and MOX use than wet storage because of higher temperatures and, consequently, higher stresses on the cladding. The ability to meet applicable regulatory limits will need to be re-evaluated for higher burnup UOX and MOX. The result of these evaluations may require, for example, a redesign of the cask heat removal and shielding systems, a decrease in the number of spent fuel assemblies that can be placed into a single storage cask, and an increased decay time in the pool prior to placement in dry storage.

— **Transportation:** Subcriticality has to be assured even under accident conditions, such as a cask drop. Higher burnup fuel may lead to significantly more hydrogen in the cladding and structure and, thus, reduced ductility. These considerations will require additional evaluation for higher burnup UOX and MOX fuels. Since MOX fuel has a similar design to UOX fuel, its mechanical behavior should not be different. The result of these evaluations may require (a) a redesign of the transportation cask heat removal and shielding systems, (b) redesign of the structural support for the spent fuel assemblies, or (c) additional cooling time prior to transport.

Kasemeyer, U., et al., "Comparison of Pressurized Water Reactor Core Characteristics for 100% Plutonium-Containing Loadings," *Nuclear Technology*, **122**, pp. 52–63 (1996).

"Comparison of Pressurized Water Reactor Core Characteristics for 100% Plutonium-Containing Loadings" documents the assessment of the possibility of using uranium-free fuel so that no new plutonium is produced. It compares the two types of Pu-burning PWR cores, 100% MOX and 100% uranium-free plutonium fuel. Both reactor-grade and weapons-grade MOX fuel was considered, resulting in the consideration of four different reactor scenarios. The all plutonium-cores reduce the plutonium mass by twice the amount as MOX-cores (~70% to ~35%). From the study of the four reactor scenarios,

conclusions were also drawn about the soluble boron concentrations, power peaking, reactivity coefficients, shutdown margins, and transient scenarios.

The report contains relevant information (core physics—soluble boron, peaking factors, and reactivity coefficients—including beta effective and void reactivity coefficients) for RG-MOX and WG-MOX, in addition to inert-matrix fuel types, and considerations for transient reactor behavior.

Kudo, T., et al., "Releases of Cesium and Poorly Volatile Elements from UO_2 and MOX Fuels Under Severe Accident Conditions," *Journal of Nuclear Science and Technology*, **44**(11), pp. 1421–1427 (2007).

The journal article "Releases of Cesium and Poorly Volatile Elements from UO_2 and MOX Fuels Under Severe Accident Conditions" was completed by the VEGA program at the Japan Atomic Energy Agency. The report documented the investigation of radioactive releases from fuel during severe accidents. The three types of fuels that were studied in the program were UO_2 fuel irradiated in a PWR, UO_2 fuel irradiated in a BWR, and MOX fuel irradiated at the ATR Fugen. The nuclear fuels were heated up to 3130 K at 0.1 MPa in a helium atmosphere. The study concluded that the cesium release in the three fuels was essentially the same. Additional results are presented concerning other nuclides at various temperatures including U, Pu, Sr, and Mo.

Leonard, M. T., et al., *Accident Source Terms for Boiling Water Reactors with High Burnup Cores Calculated Using MELCOR 1.8.5*, SAND 2007-7697, Sandia National Laboratories, Albuquerque, NM, 2007.

Accident Source Terms for Boiling Water Reactors with High Burnup Cores Calculated Using MELCOR 1.8.5 presents results of the investigation into whether the NUREG-1465 source term for BWRs requires an update. The report also investigates the assumptions used to determine the fission product release to containment in NUREG-1465. The report documented results of studies into several different scenarios of MELCOR calculations performed on two types of BWRs in hypothetical core damage accident scenarios. The report also gives recommendations for updates to the BWR NUREG-1465 source term and the basis for the changes.

Lyman, E. S., "The Impact of the Use of Mixed-Oxide Fuel on the Potential for Severe Nuclear Plant Accidents in Japan," Nuclear Control Institute presentation in Japan, 1999.

"The Impact of the Use of Mixed-Oxide Fuel on the Potential for Severe Nuclear Plant Accidents in Japan" presents the results of an NCI investigation into the use of MOX fuel and its impact on severe accidents. It compares the fuel inventories of a core with RG-MOX fuel to that of core of LEU fuel assemblies. It also compares the consequences of severe accidents involving RG-MOX (full-core and ¼-core) to the consequences of a severe accident with a LEU core. Both latent cancer fatalities and prompt fatalities are estimated, and three different plutonium release fractions are reported (0.01, 0.035, and 0.0014).

Lyman, E. S., "Public Health Risks of Substituting Mixed-Oxide for Uranium Fuel in Pressurized Water Reactors," *Science & Global Security*, **9**, pp. 33–79 (2001).

"Public Health Risks of Substituting Mixed-Oxide for Uranium Fuel in Pressurized Water Reactors" estimates the increase in risk to the public of using WG-MOX fuel in U.S. PWRs. It concluded that the

risk exceeds NRC guidelines, and, thus, the risk had to be significantly reduced before NRC approval would be granted. The paper calculates the radiological consequences to the public from PWRs with MOX fuel and compares them to PWRs with LEU fuel.

The assumption is made that severe accident probabilities are the same for LEU and MOX cores. The author's assumption is that differences in consequences are due entirely to different radionuclide inventories. The approach used is that "risk calculations are simplified" by these assumptions, and that Level 2 PRA results can simply be adjusted to the associated MOX radionuclide inventories.

A set of three source terms were used for the analysis which comprise the "large early release frequency" events—the starting composition of LEU, WG-MOX, and RG-MOX. Source terms examining beginning of cycle (BOC) and end of cycle (EOC) were examined. A range of actinide release fractions were used. The MACCS 2 code was used to estimate consequences for three severe accidents for these cores. The indicators cited were latent cancer fatalities (LCFs) within 1000 miles of the plant, prompt deaths and the average LCF risk within 10 miles of the plant. Doses were accumulated for the 1-week emergency phase. Generic parameters were used for population and for atmospheric data (one single weather scenario): a constant population density of 100 persons per square kilometer, no rain, and a wind speed of 2 m/s, and class D atmospheric conditions. Evacuation within 10 miles was assumed at containment breach. Conclusions are provided in the report, but key assumptions and statistical uncertainty ranges for the data utilized in the analyses and how that uncertainty affects the conclusions is not provided in the report.

Lundy, Dennis L., "Sequoyah Units 1 and 2—Advanced Fuel Cycle Study—Mixed Oxide Fuel Material—United States Department of Energy, Design Input Information—Evaluation of Mixed Oxide (MOX) Fuel Material on Plant Transient and Accident Analyses," TVA Memorandum, July 15, 2008.

"Design Input Information—Evaluation of Mixed Oxide (MOX) Fuel Material on Plant Transient and Accident Analyses" gives design input information for TVA Sequoyah Units 1 and 2 for the Advanced Fuel Cycle Study on Mixed Oxide Fuel Material, sponsored by the U.S. DOE. The report provides information about the radiological source term, decay heat, transient and accident response, and radiological consequences analysis.

McCoy, K., R. Morris, and B. Bevard, "Hot Cell Examination of Weapons-Grade MOX Fuel," in *Proceedings of the 2010 LWR Fuel Performance/Top Fuel/WRFPM*, Orlando, FL, September 2010.

Four lead assemblies were manufactured with weapons-grade MOX and irradiated to a maximum fuel rod burnup of 47.3 MWd/kg. As part of the fuel qualification process, five fuel rods with varying burnups were shipped to Oak Ridge National Laboratory for hot cell examination. The rods have been examined with the Advanced Diagnostics and Evaluation Platform (ADEPT); examinations include length measurements, visual examination, gamma scanning, profilometry, eddy-current testing, gas measurement and analysis, and optical metallography. Results were found to be consistent with predications and with prior experience with reactor-grade MOX fuel.

McCoy, K., R. Morris, B. Bevard, and P. Blanpain, "Performance of MOX Fuel from Nonproliferation Programs," Presented at the *2011 Water Reactor Fuel Performance Meeting*, Chengdu, China, September 2011.

As part of the fuel qualification process, four lead assemblies were manufactured with weapons-grade plutonium and irradiated to a maximum fuel rod burnup of 47.3 GWd/tonne heavy metal. Five fuel rods were selected for hot cell examination. Nondestructive exams included length measurements, visual examination, gamma scanning, profilometry, and eddy-current testing. Destructive exams completed to date include gas measurement and analysis, optical metallography of both fuel pellets and cladding, transmission electron microscopy of the cladding, radiochemical measurements (including burnup determination), gallium analysis of pellets and cladding, and hydrogen analysis of cladding. Results of these examinations are reviewed and found to be consistent with predictions and with prior experience. The results will be used to support licensing of MOX for batch use in commercial power reactors.

Murphy, B. D., *Characteristics of Spent Fuel From Plutonium Disposition Reactors Vol. 3: A Westinghouse Pressurized-Water Reactor Design*, ORNL/TM-13170/V3, Oak Ridge National Laboratory, Oak Ridge, TN, 1997.

Characteristics of Spent Fuel from Plutonium Disposition Reactors; Vol 3: A Westinghouse PWR Design gives the results of a study that investigated utilizing MOX fuel in a PWR reactor. The MOX fuel contained weapons-grade plutonium. Related to the spent fuel, the study looked at computational methods, isotopic composition of spent fuel, activity and decay heat, gamma and neutron dose rates, severe accident analyses, and criticality safety for geologic repository. The MOX fuel pin radial power profile was also investigated. Both two-cycle cases and three-cycle cases were looked at and have burnups of 35 GWd/t and 52.5 GWd/t, respectively.

Murphy, B. D., *Characteristics of Spent Fuel From Plutonium Disposition Reactors Vol. 4: Westinghouse Pressurized-Water-Reactor Fuel Cycle Without Integral Absorber*, ORNL/TM-13170/V4, Oak Ridge National Laboratory, Oak Ridge, TN, 1998.

Characteristics of Spent Fuel from Plutonium Disposition Reactors; Vol 4: Westinghouse PWR Fuel Cycle without Integral Absorber documents a study in the use of MOX fuel in a PWR reactor. This report builds on the report that looked at weapons-grade plutonium being utilized in a PWR. In particular, the transition period when the core will be different than both the MOX case and the UO_2 case is considered. Two different core arrangements were modeled because they produced the most fissile material. One fuel arrangement had 4.8% plutonium content and a burnup level of approximately 64,700 MWd/t and the other fuel arrangement considered 4.25% plutonium content and a burnup level of about 21,500 MWd/T.

Pavlovitchev, A. M., et al., *Creation of Computational Benchmarks for LEU and MOX Fuel Assemblies Under Accident Conditions*, ORNL/SUB/99-B99398V-1, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, Oak Ridge, TN, 1999.

Creation of Computational Benchmarks for LEU and MOX Fuel Assemblies under Accident Conditions presents the results of several benchmarking calculations performed on MOX fuel critical experiments. The codes used include MCU-RFFI, TVS-M, WIMS-ABBN, and TRIANG-PWR. The results presented in the report indicate agreement with K_o with differences of less than 1.5%, with most being under 1%. The authors postulated that this difference can be attributed to Pu-isotope nuclear data differences.

Pilch, M. M., K. D. Bergeron, and J. J. Gregory, *Assessment of the DCH Issue for Plants with Ice Condenser Containments*, NUREG/CR-6427 (SAND 99-2553), Sandia National Laboratories, Albuquerque, NM, April 2000.

Assessment of the DCH Issue for Plants with Ice Condenser Containments addresses the Direct Containment Heating (DCH) issue for Westinghouse PWR plants with ice condenser containments. Direct Containment Heating phenomena in ice condenser plants differ from DCH phenomena in other PWRs in that ice beds are used to suppress Design Basis Accident steam loads, AC-powered igniters to control hydrogen concentrations in the atmosphere, small containment volumes, and containment buildings with low ultimate capacities to withstand internal pressures. The probabilistic framework used addresses DCH-induced overpressure failures in the context of early containment failure modes including DCH overpressure failures, thermal failures of the containment liner, non-DCH hydrogen combustion overpressure failures, and nonexplosive steam spike over pressure failures. The study found that early containment failure probability is dominated by nonhydrogen combustion events rather than DCH events and was seen to largely depend on plant specific probabilities for station blackout (ice condenser igniter systems are not operable during station blackout events). The study showed that ice condenser plants are more vulnerable to early containment failure than PWRs with large dry or subatmospheric containments. Even though the ice condenser plants were determined to be vulnerable to blackout sequences, the weighted probability of early containment failure (i.e., averaged over all full-power events) was generally within the acceptance criteria of containment performance.

Popov, S. G., et al., *Thermophysical Properties of MOX and UO_2Fuels Including the Effects of Irradiation*, ORNL/TM-2000/351, Oak Ridge National Laboratory, Oak Ridge, TN, 2000.

Thermophysical Properties of MOX and UO_2 Fuels Including the Effects of Irradiation reviews the various relevant nuclear fuel properties that can be used in thermal hydraulics codes to study design basis or severe accidents. The fuel properties studied include the melting temperature, thermal expansion, density, heat of fusion, enthalpy, specific heat, and the thermal conductivity. The variables that influence each fuel property were studied, and they include fuel composition, temperature, porosity, fraction of the theoretical density, burn-up, and the oxygen-to-metal ratio. The properties of UO_2 and MOX fuels were found to be very similar. The main differences that were documented include MOX fuel having a lower melting temperature and a lower thermal conductivity than UO_2 fuel.

Powers, D. A., et al., *Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel*, SAND 2011-0128, Sandia National Laboratories, Albuquerque, NM, 2011.

Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel develops representative accident source terms for different reactor fuels. Representative accident source terms are calculated similar to the method used to create the NUREG-1465 source term. The three nuclear fuels for which source terms were created include high-burnup fuel in BWRs and PWRs and MOX fuel in PWRs, with an ice-condenser containment. The source terms were generated using nonparametric statistics in the MELCOR 1.8.5 computer code. The source terms developed in this study do not differ significantly from the source terms developed for lower burnup fuel and for low-enriched uranium dioxide fuel. However, they do differ from the source terms developed in NUREG-1465 because of improved knowledge of modeling of reactor accidents since its date of publication.

Reich, W. J., et al., *Impacts on Reactor Systems, Operations, Equipment, and Facilities from the Use of MOX Fuels*, ORNL/MD/LTR-140, Oak Ridge National Laboratory, Oak Ridge, TN, 1998.

Impacts on Reactor Systems, Operations, Equipment, and Facilities from the Use of MOX Fuels provides an assessment of impacts related to the complexity, technical risk, and potential licensing challenges associated with reactor operations, equipment and facility changes, and overall fresh and spent fuel handling at commercial reactors that result from converting from a low-enriched uranium fuel to MOX

fuel. Information is included regarding fresh fuel transportation and physical security issues; MOX fresh fuel handling issues; MOX fuel effects on commercial power plant systems and operations; spent MOX fuel management issues; and MOX fuel characteristics.

Ryman, J. C. and O. W. Hermann, *Characteristics of Spent Fuel From Plutonium Disposition Reactors Vol. 2: A General Electric Boiling-Water-Reactor Design*, ORNL/TM-13170/V2, Oak Ridge National Laboratory, Oak Ridge, TN, 1998.

Characteristics of Spent Fuel from Plutonium Disposition Reactors; Volume 2: A General Electric Boiling Water-Reactor Design investigates the properties of spent fuel coming out of BWRs. As part of the study, burnup and decay calculations were completed. These calculations are then used to provide data to complete other calculations involving the MOX fuel cycle. The results from the calculation involving MOX fuel are then compared to a calculation which utilized UO_2 fuel.

Sowder, A., A. Machiels, and J. Hamel, "Readiness of the U.S. Reactor Fleet for MOX Fuel Utilization," in *Proceedings of Advances in Nuclear Fuel Management IV (ANFM 2009), Hilton Head Island, SC, April 12–15, 2009*, American Nuclear Society, 2009.

Low enrichment uranium oxide fuel is used exclusively in the U.S. for commercial nuclear power production. However, the use of mixed oxide fuel in LWRs, with fissile ^{239}Pu derived from either excess weapons stockpiles or from irradiated UOX fuel, is a proven commercial technology. The use of mixed oxide fuel in light water reactors is a mature technology implemented on a commercial scale in Europe. In this report, EPRI presents a fresh look at the feasibility of MOX fuel utilization in the current U.S. reactor fleet and in newer LWR designs, assuming an adequate supply of MOX fuel and excludes policy, legislative, economic, and social considerations.

In the U.S., progress towards commercialization of MOX fuel in light water reactors ended in 1977 due to nuclear proliferation concerns. U.S. Department of Energy plans to disposition surplus weapons grade plutonium through MOX irradiation in U.S. LWRs and expanding interest in nuclear power and advanced fuel cycles indicate that MOX utilization in the current and new U.S. reactor fleet could become an option for utilities in the coming decades. The Electric Power Research Institute is reviewing the substantial knowledge base on MOX use in LWRs to evaluate the technical feasibility of MOX utilization in the existing U.S. LWR fleet and in Generation III/III+ designs under consideration by U.S. utilities. EPRI has not identified technical or licensing barriers that would preclude the use of MOX fuel in a sizeable fraction of the U.S. reactor fleet. In principle, most, if not all,

U.S. reactors could accommodate partial MOX cores. Use of higher MOX loadings (30 – 40%) can pose significant challenges that can require important plant and operational changes to address decreases in reactivity control and shutdown margins, among other factors. Core loadings of 50% or greater generally require MOX specific reactor designs. All of the newer Generation III/III+ reactors designs for the U.S. market are expected to accept 50 – 100% MOX cores, although site and plant specific features could constrain MOX use.

The total Pu in LWR fuel is generally limited to less than 12 wt %. The bulk of the fuel matrix remains as UO_2 which means the physical characteristics of MOX fuel are comparable to UOX fuel. UOX fuel derives about a third or so of its energy production from Pu over its in-core life. The differences between UOX and MOX fuel decrease with increasing fuel burnup levels.

For large fractions of MOX fuel, the shorter neutron lifetimes and smaller delayed neutron fraction associated with the presence of Pu leads to faster reactor core responses to reactivity transients. The

differences in neutronics for Pu and U result in important issues for reactor core reactivity due to overall decreased effectiveness of control and shutdown rods, soluble boron, gadolinium, and xenon. Also, there are changes in fuel and moderator temperature coefficients which result in different reactor responses that reduce safety margins, especially shutdown margins. These issues are discussed in this paper.

In addition to core operating characteristics and reactivity control, this paper discusses UOX and MOX fuel characteristics and performance, and the issues of reactor aging and materials degradation related to MOX neutron spectrum hardening. Fuel handling, operations, and on-site spent fuel management are also discussed: fresh WG and RG MOX fuel will exhibit higher dose rates relative to UOX; higher long-term decay heat from MOX fuel may delay refueling-outage core offloading; several issues related to heat generation and actinide composition have bearings on spent MOX fuel management. This paper discusses transportation and security issues too.

U.S. Department of Energy Defense Programs (DOE/DP), *Status of Mixed Oxide Fuel Utilization*, 1996.

The DOE report *Status of Mixed Oxide Fuel Utilization* gives the status of MOX fuel in 1996. It reports on the MOX nuclear fuel experience, both in the United States and abroad. The domestic summary includes information about the plutonium utilization program, the Saxton Plutonium Project, EEI-Westinghouse Plutonium Research, EPRI studies, and testing in EBWRs. The foreign MOX fuel experience includes information about programs in Belgium, France, Germany, Japan, the Russian Federation, and the UK. The report also includes information about the nuclear fuel fabrication capacity and nuclear fuel reprocessing and separated plutonium stockpiles.

APPENDIX A: BIBLIOGRAPHY OF RELEVANT MOX DOCUMENTS AND REPORTS

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