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Statement by

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on behalf of the

Natural Resources Defense Council

at Department of Energy Public Hearings

on the Draft Environmental Impact Statement

for Proposed L-Reactor Operations

Beaufort, South Carolina

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Introduction

My name is Dr. Thomas B. Cochran. I am a Senior Staff Scientist at the Natural Resources Defense Council, Inc. (NRDC). NRDC is a public interest environmental protection organization with extensive technical and policy expertise on nuclear matters, representing over 43,000 members and contributors in the United States and abroad.

I have been a consultant to numerous government agencies on matters related to nuclear energy, including the Department of Energy's (DOE) Energy Research Advisory Board (ERAB), DOE's Non-proliferation Advisory Panel, and the Energy Research and Development Administration's (ERDA) LMFBR Review Steering Committee. I currently serve on ERAB's Technical Panel on Magnetic Fusion, which was established by the Magnetic Fusion Energy Engineering Act of 1980 (P.L. 96-386). I am also a member of the Three Mile Island (TMI) Public Health Fund Advisory Board, the Nuclear Regulatory Commission's (NRC) TMI Advisory Committee, and the NRC's Special Study of Nuclear Quality Assurance. I am the principal technical expert on behalf of NRDC in the licensing proceedings for the Clinch River Breeder Reactor.

I am the author of The Liquid Metal Fast Breeder Reactor: An Environmental and Economic Critique (Johns Hopkins University Press, 1974), co-editor of the Nuclear Weapons Databook series and co-author of Volume I: U.S. Nuclear Forces and Capabilities (Ballinger, 1983, in press).

I have a Ph.D. degree in physics, an M.S. degree in physics, and a B.E. degree in electrical engineering from Vanderbilt University. I was a Health Physics Fellow under the Atomic Energy Commission's radiation training program.

While there are several important issues related to the proposed start-up of the new L-reactor, my statement will be limited to two issues: First, is the L-reactor safe -- does it meet the minimum safety standards imposed by the NRC on licensed commercial power reactors? Second, can the operation of the L-reactor be delayed long enough to incorporate needed environmental and safety technologies without risk to national security?

I. The L-Reactor Safety Issue

Turning first to the safety issue, it must be recognized that DOE facilities, such as the new L-reactor, are not licensed by the NRC. It is DOE's policy, however, to conform where appropriate to all NRC environmental and safety regulations, or, at a minimum, to meet the intent of these regulations. In DOE's own words:

> Although DOE production facilities are not subject to regulation by the Nuclear Regulatory Commission (NRC), DOE and its contractors conform to internally promulgated guides that, where appropriate, parallel or meet the intent of those of the NRC.[1]

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[1] E. I. duPont de Nemours & Co., "Safety Analysis of Savannah River Production Reactor Operation," DPSTSA-100-1, Revised Sept. 1983 (hereafter "1983 SAR"), p. 5.

For reactors licensed by the NRC, the fundamental regulations that determine the adequacy of the site and the design of the containment/confinement system for limiting exposure to the public in the event of a severe accident are embodied in 10 CFR Part 100, Reactor Site Criteria (27 Fed. Reg. 3509 (1962)). These regulations, which were developed prior to the separation of the Atomic Energy Commission (AEC) into ERDA (now DOE) and the NRC, have been used for two decades to judge the adequacy of both NRC and DOE facilities and sites. There is no debate over whether the purpose and intent of these regulations apply to DOE facilities. In fact, DOE and its contractor, DuPont, have used 10 CFR Part 100 on numerous occasions to judge the adequacy of a wide variety of containment/confinement alternatives for the production reactors at SRP.[2] Less than three years after 10 CFR Part 100 regulations

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2 Memorandum from W. S. Durant to E. C. Nelson, "Proposed Containment Shell for Building 105-C," Tech. Div. Savannah River Laboratory (SRL), DPST-64-423, Jan. 29, 1965.

Roger E. Cooper and Bernard C. Rusche, "The SRL Meteorological Program and Off-Site Dose Calculations," SRL, DP-1163, Sept. 1968.

Memorandum from S. P. Tinnes to G. F. Merz, "Airborne Activity Confinement System Base Case Design Basis Accident," Tech. Div. SRL, DPST-79-441, July 19, 1979.

Memorandum from S. P. Tinnes to G. F. Merz, "Airborne Activity Confinement System Performance First Five Hours After Reactor Accident," Tech. Div. SRL, DPST-79-555, Nov. 1, 1979

Memorandum from S. P. Tinnes to D. A. Ward, "Airborne Activity Confinement System Performance More Than Five Hours After DBA," Tech. Div. SRL, DPST-80-588, Oct. 3, 1980.

Memorandum from A. G. Evans, J. B. Price, and S. F. Petry to D. A. Ward, "Proposed Airborne Confinement System," Tech. Div. SRL, DPST-81-596, July 23, 1981.

Memorandum from W. L. Pillinger to T. V. Crawford,
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were promulgated, SRP officials noted with respect to 10 CFR Part 100 dose limits, "These values do not constitute legal limits. . . . It may be expected, however, that dose limits greater than those shown in the regulation will meet with AEC opposition."[3]

In my statement below, I will demonstrate that the L-reactor does not comply with the requirements of 10 CFR Part 100 as interpreted by the NRC in over 20 years of application. I will then explain how DOE in its draft environmental impact statement has attempted to obfuscate the L-reactor's failure to comply with 10 CFR Part 100 requirements.

A. Requirements of 10 CFR Part 100

The requirements of 10 CFR §100.11 are reproduced in Appendix A to this statement. These guidelines specify reference values for the maximum radiation dose an individual is permitted to receive at the outer boundaries of the plant and the so-called "low population zone." The reference dose values for both boundaries are 25 rem to the whole body and 300 rem to the thyroid. In assessing compliance with 10 CFR Part 100, DOE assumes that the boundaries for the SRP site and the low population zone are identical. Thus, at SRP all doses are computed at the site boundary. The doses are calculated for a 2-

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"Radioiodine Releases from Carbon Filter Desorption for Dose Calculations in Reactor SAR," Tech. Div. SRL, DPST-82-960, Oct. 29, 1982.

[3] Memorandum from W. S. Durant to E. C. Nelson, DPST-64-423, op. cit., at p. 3.

hour exposure and for a 120-hour exposure, the latter intended to cover the time period for the entire passage of the "radioactive cloud," as required by the regulation. Since the reactor locations and site boundary are already specified at SRP and thus cannot be altered, this dose assessment is used to test whether the containment/confinement technology at the production reactor is adequate, or whether it must be upgraded to meet minimum safety requirements.

B. Computation of the Maximum Site Boundary Doses

There are three procedures necessary to evaluate compliance with 10 CFR Part 100 requirements. First, the source and amount of radioactivity released to the containment by a particularly severe accident (referred to as the "source term") must be specified. Second, the atmospheric dispersion of radioisotopes, as they are carried by the wind to the site boundary, must be computed. Third, the amount of radiation absorbed by an individual at the site boundary must be computed. In each case, the methodology has been established by two decades of reactor licensing experience and regulatory guidance.

The 10 CFR Part 100 source term for light water reactors (LWRs) assumes a full core meltdown with the release to the containment building of 100% of the noble gases, 50% of the iodine (half of which is assumed to plate out within a short time), and 1% of the remaining fission products (specified in the NRC guidance document, TID 14844). We will concentrate on the noble gases and iodine since these are the most troublesome in terms of the existing L-reactor confinement technology.

An immediate question is raised: Is this LWR source term appropriate for the SRP production reactors given their differences in design? The answer is yes. As noted above, DOE has adopted the identical source term for judging the adequacy of the confinement system for existing SRP production reactors.[4] As shown below, however, DOE has responded to recent controversy by attempting to change this source term for the L-reactor, with only the thinnest of justifications.

The second step in the calculation -- atmospheric dispersion -- is calculated according to NRC Regulatory Guidelines. Since the maximum individual dose calculation is intended to be conservative, the specified meteorology has a low probability of occurrence. At SRP, less favorable meteorology and higher doses are expected only 0.5% of the time.[5]

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4 See references cited at page 3. For licensing the Clinch River Breeder Reactor, DOE and NRC have adopted the usual LWR source term (100% of the noble gases, 50% of the halogens, and 1% of the fission products) plus 1% of the plutonium in the core (NRC, "Site Suitability Report in the Matter of Clinch River Breeder Reactor Plant," NUREG-0786, June 1982, p. III-8). Even for this radically different reactor design, the assumed noble gas and iodine source terms are identical to those for the LWR and the production reactors at SRP.

5 According to the 1983 SAR, "Doses are computed by two methods. The first method computes, for the entire site (all 16 sectors), a dose (either inhalation or whole body) that would be exceeded only 5% of the time. The result is referred to as the 95th percentile value. The second method computes for each sector a dose value that would be exceeded only 0.5% of the time (a 99.5th percentile procedure). The maximum dose for all sectors is then compared to the 95th percentile dose for the

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Using data presented in the 1983 SRP Production Reactor Safety Analysis Report (1983 SAR), one can compute the maximum individual whole body and thyroid doses at the L-reactor site boundary to test compliance with 10 CFR Part 100. Table 15-4 of the 1983 SAR, reproduced in Appendix B to this statement, reports the whole body and thyroid doses associated with 1% and 3% core damage at the L-reactor. These doses are based on the assumption that 1% core damage would result in airborne release of 1% of the noble gases and tritium and 0.5% of the iodine (1983 SAR, p. 15-69). This source term value for 1% core damage need only be scaled up to 100%, or full core damage, to be consistent with the appropriate 10 CFR Part 100 source term -- release of 100% of the noble gases and 50% of the iodine. The resulting doses for the new L-reactor would be:

		Calculated Dose(rem)	
Accident	Meteorology	Whole Body 2-hour	Thyroid 120-hour
10 CFR Part 100 source term (100% noble gas & 50% iodine release from fuel)	99.5th percentile	220	1050
10 CFR Part 100 Reference Values		25	300

As can be seen, the new L-reactor does not meet minimum

whole site, and the higher of the two values is reported.

For the SRP site, the second method (99.5th percentile worst sector) gives doses (both thyroid and whole body) at the site boundary that are about a factor of two higher than the value obtained with the first method (95th percentile whole site)." Id. at p. 15-74.

safety requirements for the control of radioactivity releases in the event of a severe accident. If Congress said tomorrow, "This reactor must be licensed by the NRC," DOE would have no choice but to improve the confinement system in order to trap about 90% of the noble gases released from the reactor core after a severe accident.

C. DOE's Efforts to Mask L-Reactor Non-Compliance With 10 CFR Part 100

In response to extensive public criticism questioning the L-reactor's safety and its lack of a containment building, DOE has developed the following argument to deflect attention from the L-reactor's failure to meet 10 CFR Part 100 requirements. DOE now claims that there are no credible L-reactor accidents that could result in fuel melting of more than 3% of the reactor core and, consequently, that one should assume a design basis accident[6] and a source term which are 30 times smaller than DOE and NRC previously assumed. Based on these assumptions, DOE argues, the offsite doses associated with all credible L-reactor accidents are well within 10 CFR Part 100 guideline values. This argument simply cannot withstand scrutiny.

DOE apparently bases this argument on the fact that the SRP

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[6] The term "design basis" is used in the context of nuclear licensing to denote the range of postulated accidents for which it is required to provide protection in the form of engineered safety features systems. For purposes of 10 CFR Part 100, the NRC equates "design basis accidents" with "credible accidents." The 10 CFR Part 100 source term must be greater than that resulting from any "credible" or "design basis" accident.

emergency core cooling systems (ECCS) are currently designed to limit core melting to no greater than 1% of the fuel.[7] DOE also points to its estimates that a fuel reloading accident at SRP would result in no greater than 3% core melting (1983 SAR, p. 15-69). DOE's claims that this 1-3% fuel melting figure should be plugged into the 10 CFR Part 100 source term analysis flies in the face of both DOE's own analysis of existing SRP reactors and NRC's treatment of licensed commercial reactors.

To begin with, neither DOE nor NRC has ever used ECCS design criteria as a basis for judging the adequacy of the confinement system under 10 CFR Part 100. For light water power reactors, and historically for the DOE production reactors, NRC and DOE have assumed a full-core meltdown and the traditional 10 CFR Part 100 source term as the design basis accident for the confinement system. The 10 CFR Part 100 requirements were intended to provide a substantial additional layer of conservatism above and beyond that provided by emergency core cooling and other safety features designed to mitigate against design basis accidents. In other words, when 10 CFR Part 100 was developed, the AEC decided that, even if the plant were designed to prevent and mitigate

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[7] DOE has postulated two classes of DBAs for which the SRP ECCS should be capable of providing protection: loss-of-coolant and loss-of-circulation. J.W. Joseph, Jr., and R.C. Thornberry, "Analysis of the Savannah River Reactor Emergency Core Cooling System," SRL, DPST-70-463, Oct. 1970, p. 13. In 1970, DuPont estimated that the maximum amount of core melting for which the ECCS could be maintained was 10%. Id. at p. 17. Today, SRP establishes operating power limits designed to limit core damage from loss-of-coolant and loss-of-circulation accidents to less than 1%. 1983 SAR, pp. 15-51, 15-54.

against all credible accidents, the possibility for a much more serious, though highly improbable, accident could never be completely discounted, and therefore its consequences must be considered when siting the plant and designing the containment system.[8] As implemented, the 10 CFR Part 100 regulations state that the major accident from which the source term should be calculated has "generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products." 10 CFR §100.11(a), n. 1.[9] Thus, the history of 10 CFR Part 100 convincingly demonstrates that the regulation should not be based on ECCS design criteria.

Secondly, DOE's argument, if carried to its logical conclusion and applied to NRC-licensed reactors, would result in a complete anomaly. DOE claims that, since SRP reactor ECCSs are designed to limit fuel melting to 1%, the 10 CFR Part 100 doses should be calculated, and the adequacy of the containment tested, based on the 1% figure. Yet, reactor ECCSs licensed by the NRC are designed to permit no fuel melting whatsoever.[10] According

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[8] Atomic Energy Commission Reactor Site Criteria, Report to the Director of Regulation by the Director, Licensing and Regulation, AEC-R 2/39, Appendix D at p. 9.

[9] As noted previously, the precedent with regard to both commercial power reactors and production reactors has been to interpret "substantial meltdown with subsequent release of appreciable quantities of fission products" to mean full core meltdown with the instantaneous release to the containment or confinement system of 100% of the noble gases, 50% of the iodine, and 1% of the remaining fission products.

[10] The NRC assumes as a design basis accident a loss-of-coolant accident caused by a double-ended pipe break. Reactors must be

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to DOE's logic, NRC-licensed reactors would not even need containment buildings, since there would be no 10 CFR Part 100 offsite doses at all based on the ECCS no-fuel-melting criteria. This absurd result underscores the weakness of DOE's argument and demonstrates the need to assure sufficient conservation by basing 10 CFR Part 100 upon a substantial meltdown accident, rather than on ECCS design criteria.

Furthermore, even if DOE were somehow correct in basing the 10 CFR Part 100 analysis upon the ECCS design criterion, the 1-3% fuel melt figure is still far too low to be considered the maximum credible accident. The ECCS design criterion of not more than 1% fuel melting is based on the single failure criterion, which assumes that an accident -- e.g., a pipe break -- is accompanied by the most detrimental failure of a single active component of the system. Common cause failures, which could cause simultaneous failure of two or more active components, could cause fuel melting beyond that established as the ECCS DBA. For example, the accident at Three Mile Island Unit 2 was "beyond the design basis of the ECCS" in that there were multiple failures of active components, resulting in cladding, and possible fuel, melting well beyond the ECCS design limits.

The Three Mile Island accident points up another flaw in the DOE analysis of "credible" accidents at SRP. DOE assumes that the percent release of noble gases is directly proportional to

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designed to permit no fuel melting from this accident, even assuming the single failure criterion.

the percentage of fuel melted, e.g., 3% fuel melting results in the release of 3% of the noble gases. To the contrary, at TMI Unit 2, the percentage of the noble gas inventory released was several times the percentage of the core damaged.

In any case, the question of whether fuel melting beyond 3% is "credible" or "incredible," from the standpoint of the ECCS criteria, is irrelevant from the standpoint of the confinement system design requirements. The confinement system must meet 10 CFR Part 100 requirements. It must maintain off-site doses below 10 CFR Part 100 guideline values, assuming the release of 100% of the noble gases, if it is to achieve its "defense-in-depth" objective of limiting the risk to the public if a more serious accident, not normally considered credible, should occur. As shown above, the L-reactor simply does not meet these requirements.

As a separate matter, DOE has attempted to use probabilistic risk analyses to bolster its argument that accidents resulting in more than 1-3% fuel melting are not "credible." In essence, DOE claims that more severe accidents are not credible since the probability of their occurrence is less than one in a million (10^{-6}) per reactor year of operation. The calculations cited in the DEIS (Vol. 1, p. 4-54; Vol. II, pp. G-44 to G-48) refer to estimates made in a recent internal DuPont memorandum (J.P. Church to D.A. Ward, "Risk Estimates for SRP Production Reactor Operation," DPST-83-717, Aug. 26, 1983). This internal document, however, points out that the risk assessment will not be completed for about two years and that

> The present study should be viewed as a preliminary estimate of risk. The study is not sufficient for use as a basis for making absolute decisions about improving reactor safety. It is intended as a guide to engineering judgement in establishing priorities for the use of resources in making further improvements in reactor safety, just as the previously estimated risks and probabilities have been used in the past. Even the complete PRA will have limitations and will be used in much the same way.
>
> PRA results are inherently subject to uncertainty. In particular, PRA results cannot be expected to quantify risks from accidents or events which cannot or have not been postulated and quantified.

Id., pp. 2-3 (emphasis added).

In the DEIS, the DOE conveniently fails to mention this cautionary note,[11] and also fails to mention the caveats at the end of the DuPont document, including the following:

> The estimates of probabilities used in this study for specific accident sequences and consequences should be considered with careful regard to the assumptions made. First, the estimates of component and system failure rates or failure probabilities used in this study were not obtained by a comprehensive analysis. They are the best estimates that can be made at the present time with existing data and resources. They are judged to be reasonable. Second, the estimated rates are based upon extrapolations of experience. They do not include the probability of initiating events which could result in common failures

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11 In the Appendix of the DEIS, DOE indicates that the analysis is "preliminary" (DEIS, Vol. II, p. G-48). In the main text (DEIS, Vol. I, pp. 4-54 to 4-55), the results are presented without caveats and are presented as "fact."

> of several safety systems, and which can be postulated, but for which there is no experience base upon which to estimate probabilities.

Id., p. 16 (emphasis added).

Indeed, the failure to take into account common cause failures results in estimates of fuel melting that are likely to be several orders of magnitude too low. This renders the overall absolute probabilities meaningless for judging whether the probability of accidents resulting in more than 3% fuel melting is 10^{-6} per reactor year, as DOE would have us believe, or closer to 10^{-4} per year, or even higher.

DOE has used the same probability analysis as a partial basis for its contention that alternative containment/confinement options are not cost effective (DEIS, Vol. 1, Table 4-31, fn. d, p. 4-80). The absolute probabilities are similarly an insufficient basis for this contention.

The DOE comparisons of the cost effectiveness of alternative containment/confinement options (DEIS, Vol. I, Table 4-31) contain even more fundamental errors that render them useless. It is perhaps useful to mention several of these errors, although I do not intend to discuss them in detail in this statement.

(1) It is inappropriate to include a production loss of $150,000 per reactor-day without including offsetting operating cost that would not be incurred.

(2) The estimated man-rems averted do not include exposures:

(a) to persons exposed on site,
(b) to persons exposed at a distance greater than 80 km,
(c) to organs other than whole body, e.g., thyroid, and bone,

(d) associated with fuel-melting beyond 10% of the core.

Recognizing these inherent deficiencies, the NRC has decided that this cost-benefit approach should under no circumstances be used as a substitute for existing regulatory requirements. These requirements include ensuring compliance with 10 CFR Part 100, performing adequate site selection, and ensuring that the containment/confinement system is adequate for the protection of public health.

In sum, the L-reactor, as presently designed, is simply unsafe. It does not meet the minimum standards for design of a containment/confinement system to protect the public health in the event of a severe accident. Following the recent controversy over the adequacy of the L-reactor confinement system, DOE has attempted to lower its safety requirements -- reducing the requirements for confining noble gases by a factor of 30 -- rather than improve the confinement technology.

Simply stated, DOE believes its reactors should be held to the nuclear regulatory requirements of the Truman and Eisenhower administrations rather than today's standards. We disagree.

II. The National Security Issue

I will now turn to the national security issue. Here, the central question is whether DOE can safely defer the restart of the L-reactor in order to incorporate the technologies needed to meet today's minimal environmental and safety standards. Can we have both a safe and clean environment and adequate national security, or must the former be sacrificed for the latter, as DOE would have us believe?

In the DEIS (Vol. I, Chapter 1), DOE's emphasis on the "need" issue has been in terms of whether the L-reactor should be restarted at all, rather than the less demanding question of whether restart of the L-reactor can be deferred. A 36-month delay in L-reactor operations is ample time to upgrade the environmental control and safety systems. This period would permit installation of four of the five confinement/containment alternatives (DEIS, Vol. 1, p. 4-80), and would also permit the installation of mechanical draft cooling towers (DEIS, Vol. 1, p. 4-95). The cost of a 36-month delay in terms of foregone plutonium production is approximately 1.5-1.75 MT of plutonium. Thus, the central question here is whether 1.5-1.75 MT of foregone plutonium production is a threat to national security, or, alternatively, whether this amount (or some fraction thereof) can be supplied by other production initiatives without incurring a shortage of plutonium "needed" for nuclear weapons production.

To place this issue in perspective, it should be noted that the U.S. nuclear weapons stockpile currently contains some 80 to 90 metric tons of plutonium and 600 to 700 metric tons of highly enriched uranium. It is incredible to think that a 2 percent change in the plutonium inventory would be detrimental to national security. Certainly, we cannot estimate the number of Soviet warheads or weapons material production to that level of accuracy.

Setting this argument aside, there is strong evidence that restart of the L-reactor can be delayed for at least 36 months without incurring a shortage in plutonium to meet DOE projected weapon requirements.

A. Would a Near-Term Shortage of Plutonium Be Incurred By a Delay in Start-up of the L-reactor?

First, the DEIS fails significantly to give special consideration to a short-term delay in L-reactor operation and the shortages of materials, if any, that this delay would incur, even without alternative production options. The relevant questions that must be asked are: Would a near-term shortage occur, and, if so, could the alternative production options eliminate it?

When the 1981-83 Nuclear Weapons Stockpile Memorandum (NWSM) was signed by President Carter in October 1980, DOE projected that, unless the new production initiatives were implemented, there would be a shortage of plutonium in 1985 or shortly thereafter. With the implementation of several planned initiatives, including the restart of the L-reactor (DEIS, p. 1-3), a plutonium shortage was not projected to occur prior to the early 1990s. DOE indicates that "the increased defense nuclear material requirements . . . have been reaffirmed in subsequent Stockpile Memoranda" (DEIS, p. 1-2), but that "Congress has delayed or failed to fund certain nuclear weapons systems" (DEIS, p. 1-2). The effect has been to eliminate the shortage previously projected to occur in the early 1990s. In my view, foregoing plutonium production in the L-reactor for 36 months, even if none were made up through alternative near-term production initiatives, would not create near-term shortages. In the long term (after 1990), shortages that might otherwise appear can be made up by a variety of production initiatives, several of which are identified below.

DOE apparently does not dispute this view. Rather, DOE simply asserts that "none of the [alternative] production options, or combinations of options, would provide sufficient material in time to fully compensate for the delay or loss of L-Reactor production" (DEIS, p. 1-6). But this is not the relevant question. As stated above, the questions are: Would a near-term shortage occur, and, if so, could the alternative production options eliminate it.

B. The Recent Delays in Weapons Systems Have Significantly Reduced the Near-Term Requirements for Plutonium.

This can be seen by comparing the weapons requirements set forth in the Carter FY 1981-83 NWSM against today's requirements.

The FY 1981-83 NWSM, signed in October 1980, included a significant increase in warhead production and was the impetus for materials production initiatives. Included in this NWSM were:

- the first firm requirements for 700 W84 and W85 warheads for Pershing II and Ground-Launched Cruise Missions,
- some 2000 MX missiles warheads planned for a 200-missile force,
- sufficient W76 Trident I warheads (5,520) for backfit into 12 Poseidon submarines and 15 new Trident submarines,
- 1200 W-70-3 Lance and W79 8-inch nuclear artillery

warheads built as fission warheads with the technical ability to be shifted to enhanced radiation yields,

- 460 W80-0 Sea-Launched Cruise Missile warheads,
- 3,394 W80-1 Air-Launched Cruise Missile warheads, and
- 1000 W-82 155-mm fission artillery warheads.

The FY 1983-88 NWSM signed by President Reagan in November 1982 made significant changes to its early assumptions, which were similar to the Carter Administration:[12]

- only 1000 MX warheads would be built for 100 MX missiles,
- W76 Trident I warhead production would be cut to 3840 in the short term, with a shift to Trident II production in time for fitting the ninth Trident submarine (1989),
- the W70-3 Lance and W79 8-inch nuclear artillery warheads would be built as enhanced radiation warheads,
- 758 rather than 460 W70-0 Sea-Launched Cruise Missile warheads,
- a significant reduction in near-term W80-1 ALCM production from 3,394 to 1,739 with shift to the Advanced Cruise Missile, and
- a shift from fission to enhanced radiation yield for 1000 W82 155-mm warheads.

Significant reductions in nuclear material requirements have resulted from Reagan's decision to shift the MX warhead from the

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[12] Nine warhead types continue in production during 1983:
- the B61-3/4 bomb,
- the W76 Trident I warhead,
- the W79 enhanced radiation artillery warhead,
- the W80-0-0 Sea-Launched Cruise Missile warhead,
- the W80-1 Air-Launched Cruise Missile warhead,
- the B83 Modern Strategic Bomb,
- the W84 Ground-Launched Cruise Missile warhead,
- the W85 Pershing II warhead, and
- the W87 MX warhead.

W78 design to the W87. In addition, DOE has considerable flexibiltiy in the rate of retirement of old warheads.[13] This is the primary source of material for new weapons production.

The 1983-88 NWSM also included a number of new retirement initiatives, including retirement of B-52Ds and accelerated retirement of B52Gs (with the reduction in bomb needs), retirement of the Titan II, and accelerated retirement of Polaris. The retirements traditionally account for a large proportion of nuclear materials for new warheads. By the end of the decade, some nine warhead types (W25, B28, W31 Nike Hercules, W33, B43, W50, B53, and W76) will be retired either in part or in full.

C. Alternative Plutonium Production Initiatives Are Available to Make Up for a Potential Loss of Some 1.5-1.75 MT of Plutonium Within the Three-Year Period the L-Reactor Is Deferred.

Since 1981, DOE has exceeded its plutonium equivalent production goal. Consequently, part of the 1.5-1.75 MT Pu alternative production requirement has already been met. We estimate that DOE has surpassed its planned production goal at Savannah River by about 0.5 MT in FY 1982-83. At Hanford, the conversion of the N-reactor to the weapon-grade mode of operation was completed in FY 1982, approximately five months ahead of

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13 Two significant restraints exist in retiring warheads when scheduled: warhead retirements contingent on replacements (particularly when lack of Congressional funding slows down replacements) and double sets of warheads necessary when enhanced radiation replacements for fission warheads (W70-3, W79, and W82) are kept in the U.S. and a full set of overseas deployed warheads are also kept.

schedule, providing some 0.23 MT of additional plutonium. Thus, the makeup needed from alternative sources is only on the order of 0.8-1.0 MT.

D. Other Alternatives to L-Reactor Operation

(1) Mark-15 Cores. The use of Mark-15 cores could boost plutonium production by at least 25% per reactor. If such cores are installed in two operating SRP reactors, weapon-grade plutonium production (with blending) could be increased by 0.375-0.475 MT per year. Plans exist to install Mark-15 cores in one reactor in late FY 1985 or as late as August 1986. Accelerating introduction of the Mark-15 cores by one year could provide approximately one-half of the plutonium makeup required.

(2) Production of 5% Pu-240 Plutonium at the N-reactor. The shift from 6% to 5% Pu-240 production would produce greater quantities of plutonium than a 10% increase in N-reactor power (DEIS, pp. 2-5, 6). Such a shift could therefore increase plutonium production through blending by about 90 kg/yr, or some 0.27 MT over the next three years.

(3) Restart of the Purex Reprocessing Plant at Hanford. DOE now plans to restart the Purex Reprocessing Plant at Hanford in April 1984 to process stored and new N-reactor spent fuel to recover both fuel-grade and weapon-grade plutonium. Restart of the Purex plant three months earlier would provide an additional 100 kg of plutonium per month, or 0.3 MT total.

E. Summary: Production Options and Proposed Action

We take issue with the DEIS claim that no combination of production options can fully compensate for the loss of material that would be produced by the L-reactor if restart is delayed (DEIS, p. 2-1).

As noted above, DOE has given short shrift to its discussion of the combination of production options by failing to examine quantitatively the effect of a 36-month restart delay. The combination of the following alternatives can make up the 1.5-1.75 MT Pu-equivalent loss prior to a shortage developing in the Pu stockpile:

(a) Excess Pu already obtained by exceeding previously planned production goals.

(b) Operating N-reactor to produce 5% Pu-240 product.

(c) Accelerating Purex by 3 months.

(d) Accelerating Mark-15 core by 1 year.

This combination of alternatives would permit much needed improvements in L-reactor environmental control technology while still meeting defense nuclear material needs.

This concludes my statement. NRDC will be submitting to DOE more extensive comments on the L-reactor DEIS prior to the close of the comment period in two weeks. Thank you.

APPENDIX A

Requirements of 10 CFR §100.11

10 CFR §100.11 states, in relevant part:

(a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire peirod of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

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1 The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

2 The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

<u>APPENDIX B</u>

TABLE 15-4

Calculated Radiation Dose to a Person at the SRP Site Boundary Following Four Specific Accidents

Accident	Operating and Meteorological Conditions*	Calculated Dose, rem Whole Body (2 hr)	Thyroid (2 hr)	Thyroid (120 hr)
Reference values for reactor siting in 10 CFR 100.[3]		25	300	300
D_2O Spill	Typical	0.007		
	Very Unlikely	0.14		
Discharge Mishap (one fuel assembly melts)	Typical	0.0038	0.0078	0.018
	Very Unlikely	0.055	0.12	0.29
Misloading Criticality (3% core damage)	Typical	0.39	0.48	1.4
	Very Unlikely	6.6	11.1	31.5
Hypothetical LOCA (1% core damage)	Typical	0.13	0.16	0.46
	Very Unlikely	2.2	3.7	10.5

* Typical conditions are 2500 MW reactor power, average (50%) meteorology, and 19-month service age carbon filters (carbon filter age is discussed in Section 15.3.2.2). Very unlikely conditions are maximum anticipated reactor power of 3000 MW, very unfavorable meteorology as specified in NRC Regulatory Guide 1.145 (95% site, 99.5% worst sector), and 19-month aged carbon filters. Values shown are maximum for any of the P, L, K, and C Reactors. The core inventory of tritium is included in the whole body calculations.

APPENDIX C

Evolution of the Confinement Technology at SRP Production Reactors

The production reactors at SRP were constructed in the early 1950s. The L-reactor, the third of five, began operating in July 1954. SRP originally controlled airborne radioactive releases by dispersion via tall stacks (DEIS, Vol. II, p. J-1). SRP also relied on the fact that the site extended over 300 square miles, thus permitting greater dispersion of radioactivity prior to reaching the site boundary. The L-reactor is some 9 km from the SRP site boundary (DEIS, p. 2-10). In 1958, the AEC's Advisory Committee on Reactor Safeguards (ACRS), after performing an extensive review of the SRP safety philosophy, concluded:

> The buildings in which the SR reactors are housed do not possess any significant containment features, such as those now being provided for power reactors located in more populated areas. In the event of a serious accident that would breach the reactor tank and shield, the building shell in itself could not be expected to provide a third line of defense of any consequence on restraining the volatile fission products.
>
> It was recommended that the Du Pont Company explore alternative paths toward obtaining a higher degree of confinement that is now in effect.

DEIS, Vol. II, p. J-7.

Also in 1958, the capacity of the SRP primary coolant pumps was approximately doubled (from 78,000 gpm to 150,000 gpm) which permitted a doubling of each reactor's power from about 1000 megawatts thermal (MWt) to approximately 2000 MWt (DEIS, Vol. II, pp. J-3 and J-6). Since the fission product inventory of noble gases and iodine is proportional to reactor power, this change effectively doubled the magnitude of the consequences of a serious fuel meltdown accident. Since 1958, the power level of the production reactors has been further increased, and the L-reactor is currently expected to operate at 2350 MWt* (DEIS, Vol I, p. 2-14).

In 1960-61, in response to the ACRS criticism, SRP began a major confinement system improvement project. This system would remove airborne contamination, particularly Iodine-131, through moisture separators, particulate filters, and halogen absorbers (carbon) in the process area ventilation exhaust stream (DEIS, Vol. II, p. J-7). This filtration system, while lowering the thyroid dose from halogen releases, was, however, incapable of removing noble gases, the primary contributors to the whole body dose.

In the 1950s, there were no criteria specifying the degree of site isolation or reactor containment considered desirable for mitigating the consequences of severe reactor accidents. In 1962, after extensive public comment, the AEC promulgated the 10 CFR Part 100 site suitability regulations for licensed power reactors. Throughout the remainder of the 1960s, DuPont and the AEC examined a number of alternative containment/confinement proposals. Although some of these proposals, if adopted, would

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* The highest power level achieved at SRP was 2915 MWt.

bring the SRP production reactors into compliance with 10 CFR Part 100, they were rejected because of their expense.

Improvements were made in the confinement system in the 1970s, including the installation of a Confinement Heat Removal System to avoid overheating the filter system in the event of a full core meltdown. This systems was needed because overheating the filters would reduce their retention capacity and cause desorption of the collected iodine (DEIS, Vol. II, p. J-13), thus defeating the purpose of the filters. This and other improvements, however, offered no reduction in the whole body dose due to accidental noble gas releases.