UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD '82 SEP -9 P2:14

In the Matter of	Docket No. 50-142 OL
THE REGENTS OF THE UNIVERSITY OF CALIFORNIA	(Proposed Renewal of Facility License)
(UCLA Research Reactor)	

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

CBG MOTION FOR SUMMARY DISPOSITION AS TO CONTENTION XIII

(Special Nuclear Materials License)

I. THE MOTION

Pursuant to 10 CFR 2.749 and the Board's Order of July 26, 1982, the Committee to Bridge the Gap (CBG) respectfully moves the Atomic Safety and Licensing Board for summary disposition as to Contention XIII or, in the alternative, partial summary disposition thereof.

In support of its Motion, CBG submits numerous items of documentary evidence as well as the declaration of Dr. David Hafemeister, an expert in nuclear non-proliferation matters as they relate to the use of Highly Enriched Uranium (HEU) by research reactors. These materials attest to the incontrovertible material facts set forth herein, to wit: that the amount and enrichment of Special Nuclear Materials requested by UCLA in its license application are excessive, that the proposed license activities can be performed with far lower amounts and enrichment, and that grant of the license in the amounts and enrichment requested would entail unnecessary risks to public health and safety and the common defense and security. In addition, it will be demonstrated that there is no genuine dispute about the failure of Applicant to include in its license application the information required

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by 10 CFR 70.22(a)(7) and (8) and 10 CFR 70.24(a)(1), (2) and (3), particularly with regards criticality accident protection, mitigation, monitoring, and response.

On the basis of the material facts attested to herein, and the admissions of the other parties, included in their answers to interrogatories and related documents, CBG is entitled as a matter of law to a ruling in its favor on Contention XIII, as no genuine dispute exists which would necessitate a hearing. Should the Board determine that certain residual matters as to the overall Contention remain in dispute, CBG respectfully requests that the Board grant partial summary disposition as to those material facts not in dispute.

II. THE CONTENTION

A. Background

The Applicants in this proceeding, the Regents of the University of California, have applied for a license to operate their research reactor for an additional twenty year period. Included in that Part 50 application for a facility license was a Part 70 request for a license for Special Nuclear Material [1/] to be used in conjunction with the reactor. At page 5 of the application for renewal of the facility license, Applicant states:

"Other licenses applied for in connection with this facility:

Special Nuclear Material: (1) 4700 gms U-235 (irradiated)
(2) 4700 gms U-235 (fresh)
(3) Pu-239 as a 2 Curie, Pu-Be neutron source"

1/ Special Nuclear Materials, or SNM, are defined primarily as plutonium and as uranium enriched in either the isotope 233 or 235. These are the primary materials capable of a sustained fission chain reaction. See 10 CFR 70.70.4(m) and Section 11.aa. of the Atomic Energy Act of 1954, as amended.

The 9400 grams of Uranium-235 are to be in the form of metallic uranium of 93% enrichment. The 2 Curie Plutonium-Beryllium neutron source represents approximately 32 grams of Plutonium-239.

Among the matters to be decided by the Atomic Safety and Licensing Board which has been established to rule on UCLA's application, in addition to whether to grant an operating license for the requested period, is whether the Applicant has adequately demonstrated that the requested license for nearly 10 kilograms of highly enriched uranium and over 30 grams of plutonium should be granted. Paramount in this decision is a determination whether the proposed license can be granted without undue risk to public health and safety and the common defense and security. For, as the Congress of the United States found in mandating that the Commission regulate these materials:

> The processing and utilization of source, byproduct, and special nuclear material must be regulated in the national interest and in order to provide for the common defense and security and to protect the health and safety of the public.
>
> Section 2.d. of the Atomic Energy Act of 1954, as amended, 42 U.S.C. sec. 2012

This is especially essential with regards Special Nuclear Materials, due to their unique hazards if misused.

It is perhaps the chief irony of this age that the special materials which, when fissioned in a controlled fashion inside a nuclear reactor, can produce such useful power as well as research and thereapeutic application, can also be used to make a nuclear weapon. While the fuel generally used in nuclear power plants cannot, without considerable enrichment or reprocessing, be used directly in a nuclear weapon, that is not, as shall be discussed infra, the case for the kind of highly enriched SNM requested by UCLA. The threat to common defense and security, as well as public safety, consequent to such material falling into the wrong hands is obvious.

An additional hazard attendant to use of SNM is that, in addition to being able to go "critical" (that is, sustain a chain reaction) in a controlled situation inside a nuclear reactor or explosively in a nuclear weapon, SNM can, if accidentally placed in the right configuration, go critical in unintended settings. Incidents such as these are called "criticality accidents," involve small unintentional nuclear explosions resulting in intense localized radiation, and require special procedures and care to prevent. Approximately thirty such "criticality accidents" have occurred in the United States, roughly one per year of the nuclear era, resulting in six deaths and numerous other radiation injuries, as will be discussed below. For this reason, the Commission's regulations require applicants who wish to possess more than a relatively small (greater than 700 grams) quantity of SNM[1] to provide detailed information as to how they intend to prevent and deal with criticality accidents.

Finally, the SNM itself poses a hazard due to its radioactive nature as opposed to its capability of fissioning. In particular, Plutonium-239 is one of the most toxic materials known (about 20,000 times more toxic by weight than cobra venom or potassium cyanide[2]; permissible levels are measured in billionths of billionths of Curies.[3]) Release of such material in the form of an aerosol of finely divided particles (as in a fire or through theft of the material for a radiological weapon) could have extremely serious public health consequences, consequences which would be environmentally of great longevity, given the 24,400 year half life of Pu-239. Therefore, the material is regulated carefully; the Commission is not to permit its use unless an applicant can demonstrate that its use of the material will not be inimical to public health and safety.[4]

2/ See, e.g., Theodore B. Taylor and Mason Willrich, Nuclear Theft: Risks and Safeguards, a Report to the Energy Policy Project of the Ford Foundation, Ballinger Publishing Company, Cambridge, Mass., 1974

3/ See 10 CFR 20, Appendix B

4/ 10 CFR 70.23(a)(3) and (4)

In sum, the Atomic Safety and Licensing Board has before it, in addition to UCLA's request to be permitted to operate its 22-year-old reactor until the turn of the century, a related request to be permitted to possess and use approximately 10 kilograms[5/] or 22 pounds of weapons-grade uranium (93% enriched) and approximately 32 grams of Plutonium-239. The ASLB must determine whether the Applicant has provided reasonable assurance that grant of the requested materials will not be inimical to common defense and security and public health and safety. CBG has placed that matter at issue in this proceeding. Contention XIII, subject of the instant motion, focuses directly on the SNM License Application, asserting that it fails to provide the information required by the regulations and, more importantly, that the amount and enrichment of SNM requested pose unnecessary proliferation and health and safety risks. Other concerns raised in other CBG contentions (for example, that the security plan to protect the requested material is inadequate) would be, at least in some measure, mooted by a Board determination that the amount and enrichment of SNM requested by UCLA are in excess of that reasonably needed to perform the proposed licensed activities. Contention XIII is described below.

B. Contention XIII

The Contention as admitted states as follows:

> The information which Applicant has provided regarding the special nuclear materials license is inadequate to meet the requirements of 10 CFR 70.22(a)(7) and (a)(8) and 70.24(a)(1),(2), and (3). Furthermore, the enrichment level requested and the quantity requested of SNM are excessive and thus pose an unnecessary threat to public health and safety.

5/ UCLA has actually requested slightly over 10,000 grams of 93% enriched uranium, of which 9400 grams is therefore to be U-235.

The first part of the contention alleges that important information, required by the regulations and necessary for a favorable decision to grant the requested license, is missing from the application, particularly with regards procedures and equipment to prevent, mitigate, monitor and respond to criticality accidents. Absent such information and, more importantly, absent adequate procedures and equipment, reasonable assurance cannot be given that grant of the requested license would not be inimical to public health and safety due to a criticality accident.

The second part of the contention alleges that the U-235 and Pu-239 requests are excessive, that UCLA doesn't need the amounts and enrichments asked for, and that because of the unique hazards associated with plutonium and weapons-grade uranium, the application should not be granted in the amounts and enrichment requested. The threats to public health and safety from detonation of a clandestine fission explosive produced with uranium that could be diverted or stolen from the UCLA facility are extraordinarily grave; the increased radiological dangers associated with increased criticality accident risks and hazard from accidental or intentional release of plutonium are also of concern.[6/] These hazards would be substantially reduced or eliminated if UCLA were to perform its desired activities with less potentially dangerous SNM levels. As UCLA has reduced its SNM holdings to roughly half the quantity it requests in its application, and since UCLA has for many years used a Radium-Beryllium neutron startup source for the reactor as opposed to the requested Plutonium-Beryllium source identified in the application, grant of summary disposition on those portions of the contention would

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6/ By "unnecessary threat to public health and safety" in Contention XIII, CBG refers to the range of threats from harm from detonation of a clandestine fission explosion to the lower order threats--though still worrisome--from criticality accidents and dispersion of Plutonium by accident or intent. For purposes of clarity in this motion, those separate threats to the public will be discussed separately.

merely bring the application into conformance with the status quo at the facility.

The aspects of the Contention discussed in this Motion will be in the following order: (1) criticality accident information, (2) the need for 32 grams of Plutonium, (3) the need for 9400 grams of U-235, and (4) the need for 93% enriched uranium. By so doing, it should not be inferred that these matters are of equal importance. While real public health and safety concerns exist as to all four aspects, the nuclear weapons proliferation threat occasioned by unnecessarily large quantities of weapons-grade uranium by far predominates.

III. CRITICALITY ACCIDENT PREVENTION INFORMATION

A. What is Meant by the Term "Criticality Accident"

Special Nuclear Materials are unique in their ability to release enormously large quantities of energy in astonishingly small periods of time. This is due to the fact that the nuclear chain-reaction can increase in magnitude exponentially, all in an exceedingly small part of a second. For example, in an atomic bomb, energy equivalent to hundreds of tons of high explosives can be released in a period measured in millionths of a second if just a few kilograms of highly enriched uranium are rapidly brought together or imploded and if one neutron is present at the start to begin the reaction.

The amount of SNM necessary to just barely sustain a chain reaction is the critical mass. If more than one critical mass is assembled under the right conditions, the reaction goes "super-critical", i.e. power increases exponentially until something--intervention of control rods in a normally operating reactor or explosive disassembly of the device

in a bomb--makes the assembly go subcritical again. In the interval, lasting perhaps only milliseconds, considerable energy can be released. It is thus obvious that unintended supercriticality is strongly to be avoided.

Unintended supercriticality is often called a "criticality accident." It occurs when two or more subcritical masses of SNM are accidentally brought together in the right configuration and with the right conditions (moderation, reflection, etc.) so that a chain reaction occurs where none was planned.

These accidents can be very dangerous because they are often associated with intense neutron and gamma radiation bursts and even on occasion small explosions. There have been at least six immediate deaths from such accidents and scores of serious radiation injuries from doses in the hundreds of rads, causing the Hiroshima-type acute radiation syndrome.

Criticality accidents can occur inside a reactor or outside. Because of the capability of nuclear reactions to increase in power exponentially in times considerably shorter than a person can respond, nuclear reactions must be carefully controlled if they are to be used safely in peaceful applications. In a reactor this is done by "reactivity" controls (reactivity is essentially that which makes a reactor react, something like how much horsepower one has "under the hood") such as neutron-absorbing control rods. These devices keep the nuclear reaction from getting out of control, something similar to brakes on a car except that the reactor "goes" by letting up on the brakes rather than stepping on the gas. Occasionally the brakes fail or someone makes a mistake and lets up on them at the wrong time and the fission process runs wild,

power goes from zero to millions of watts faster than you can blink your eye, and anyone unfortunate enough to be in the same room at the time gets a sizeable radiation dose capable of causing radiation sickness or death within a few hours or days. Such in-reactor criticality accidents occurred in 1952 at Argonne National Labs, 1961 at Idaho Falls, and 1958 at Vinca, Yugoslavia and resulted in some tragic deaths and injuries from the intense radiation fields generated.5/

Such accidents can also occur outside reactors, and have. Whenever SNM of greater than a certain enrichment and quantity is handled, special procedures and equipment must be employed to avoid accidentally bringing enough SNM into a proper configuration that causes it to go super-critical. Two fatalities occurred at Los Alamos within a year of each other when; in one case the individual dropped a a reflector brick and in the other a screw driver used as a wedge to hold up part of an assembly slipped. In both cases a characteristic "blue glow" was observed and the victims died within a month from the intense radiation exposure.

Because of the dangers of criticality accidents attendant whenever more than a few hundred grams of SNM are handled, the Commission requires applicants for such materials to demonstrate that they can and will take the necessary precautions to prevent such accidents occurring and will be able to respond appropriately if they do occur.

B. The Legal Requirements

10 CFR 70.22(a)(7) and (8) require an applicant for an SNM license to include in their applications "description of equipment and facilities which will be used by the applicant to protect health and minimize danger to life or property (such as handling devices, working areas, shields... criticality accident alarm systems, etc.)" and "/P/roposed procedures to

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5/ See, for more details about criticality accidents, WASH 1192, Operational Accidents and Radiation Exposure Experience Within the USAEC, portions of which are attached.

protect health and minimize danger to life or property (such as procedures to avoid accidental criticality. . .post-criticality accident emergency procedures, etc.)". And 10 CFR 70.24(a) requires all licensees authorized to possess more than 700 grams of U-235 of greater than 4% enrichment to have a criticality monitoring and alarm system meeting the specifications set out therein, as well as emergency procedures for each area in which such SNM is handled or stored for response to a criticality accident.

These requirements are matters of law; they are especially important for the Applicant in this case to obey, given the fact that it has far more than 700 grams of U-235 and far higher enrichment than 4%, and given the existence of considerable quantities of moderating and reflecting materials in rooms where SNM is stored and used (e.g. heavy water and graphite). Yet the information is lacking from the application.

The Required Information is Lacking

In interrogatories dated April 20, 1981, CBG asked the Applicant the following question (interrogatory 3 as to Contention XIII):

Precisely on what pages of the Application does Applicant provide the information required by 10 CFR 70.22(a)(7) and (a)(8) and 70.24(a)(1), (2), and (3)?

The Applicant responded on May 20, 1981, (page 135) as follows:

10 CFR 70.22 (a)(7): Appendices II and V, for examples pages III/5-15, V/3-4. 10 CFR 70.22(a)(8): Appendix V, for examples, page V/3-8, and Appendix IV. 10 CFR 70.24(a)(1), (2), and (3): None.

Applicant thus admits than none of the information identified in 10 CFR 70.24 (a)(1), (2), and (3) is found in the application. Furthermore, the page cited by Applicant with regards the 70.22 information, III/5-15, merely says that fuel loading is directed by a reactor operator--not what procedures and equipment are employed to prevent accidental criticality.

Page V/3-4 deals with general reactor room monitors; no mention is made of criticality monitors, nor of coverage of areas other than the reactor room. Information necessary to judge compliance of the monitors with 10 CFR 70.24 is not provided, as Applicant itself admits.

Page V/3-8 simply describes the fuel loading. Appendix IV formerly mentioned procedures for dealing with radiation accident cases, without specific mention of criticality cases, but after NRC Staff questioned tha assertion in the original Appendix IV (page C-1) that "the individual who has received whole or partial body radiation and may have received a lethal dose of radiation, but is no hazard to attendants, other patients or the environment," even that minimal reference to victims of direct radiation was removed. (The Staff rightly pointed out that a victim of neutron radiation, as in a criticality accident, would be radioactive himself, due to activation of sodium in the blood, gold fillings, and the like, and could be a hazard to attendants, etc.) No reference whatsoever to means for coping with criticality accidents is found in the revised Emergency Plan, the new Appendix IV.

In response to CBG interrogatory number 5 of the set identified above, which asks "What specific means are employed by Applicant for monitoring for accidental criticality of irradiated fuel?" the response was simply, "None."

Given Applicant's admission in interrogatory responses that none of the information regarding 70.24 criticality monitoring and response is in the application, and the admission that no means are employed for monitoring for accidental criticality of irradiated fuel (of which the requested license is for 4700 grams, far in excess of the 70.24 threshhold of 700 grams), there appears no dispute as to the material facts and CBG is entitled as a matter of law to a favorable ruling on that part of the contention that the identified and required information is lacking.

That this result is warranted is further supported by Staff's answers to interrogatories 231-233 by CBG as to the Safety Evaluation Report (found at page 14 of Mr. Bernard's affidavit answering the interrogatories). These three questions asked for various information about the potential for and means of preventing accidental criticality in the storage cabinet in which the fresh fuel is kept. Staff's one word answer in each case was: "Unknown."

Complete information about criticality protection at the UCLA facility is not in possession of Staff, it is not included in the Application, and in absence of that information being fully provided in the SNM license request, reasonable assurance that adequate protection, detection, and response measures will be taken is impossible.

IV. PLUTONIUM SOURCE UNNECESSARY

On October 3, 1960, the Atomic Energy Commission granted UCLA's request for a license to possess 3.350 kilograms of U-235 and 32 grams of Plutonium-239, the latter for use as a neutron startup source for the reactor. In early 1961, UCLA requested the Commission amend its license to replace the Plutonium source with a far smaller (and less hazardous) Radium source. As the AEC Hazards Analysis of June 28, 1961, described the proposed amendment:

> The applicant proposes to replace the 2 curie Pu-Be source with a 10 millicurie Ra-Be source. The present Pu-Be source has been determined to give a much stronger indication than required for safe startup. Both types of sources have been utilized successfully in research reactors; we anticipate that no additional hazard will result from the replacement of Pu-Be source with the Ra-Be source.

Despite the fact that the amendment was granted, and UCLA has used Radium startup sources ever since, it has continued to carry on its license for the reactor the authority to possess up to 32 grams of Pu-239 as a neutron source.

Note that the same application which at page 5 requests 2 Curies of Plutonium as a startup source indicates at page III/6-5 that the facility now uses a 6.6 milliCurie Radium source instead. The Radium source in use currently is many, many orders of magnitude less hazardous than the requested Plutonium source which it appears the reactor does not need; in fact, the above-cited AEC analysis would appear to indicate that reactor operations themselves are safer without the very strong indication provided by the Pu-Be, which is why UCLA discontinued using such a source in the first place.

UCLA may have in mind use of a Plutonium source for some purpose other than use related to the reactor and the activities licensed under reactor license R-71. In fact, UCLA has at various times had Special Nuclear Materials licenses that permitted use of Plutonium sources for uses other than the reactor. But these were granted on SNM licenses separate from the reactor's license. The SNM license for the reactor is for SNM for the reactor. A Plutonium source is no longer used for the reactor, hasn't been used for years, is not needed, and would be an unnecessary public health and safety hazard. If the University is attempting to hold onto a Plutonium source, or at least license for such a source, when it is no longer used for the purpose for which the license was granted and for which the renewal application has been made, then the University is being less than frank with the Commission.

In light of UCLA's own request to the Commission in 1961 to be able to use a radium source as its reactor's neutron source, the Commission's concurrence at the time that the radium source was preferable, and twenty years' operating history with the radium source, and in light

of the undisputable hazards associated with use of plutonium, and in absence of any compelling reason why the radium source should cease to be used in favor of a return to plutonium, CBG respectfully suggests that there are no material facts in dispute about its contention that the requested license for 32 grams of Plutoniumis an unnecessary threat to public health and safety and should, as a matter of law, be denied.

V. AMOUNT OF U-235 REQUESTED IS EXCESSIVE

A. History

On May 30, 1959, UCLA applied to the Atomic Energy Commission for a construction permit for a training reactor facility.[6] That Application included a request for 4.0 kg of 90% U-235 and 2 Curies of Pu-239 as a startup source. UCLA indicated that 4 kg was 660 grams more than it needed and would return the excess after fuel fabrication on or about December 30, 1959.[7] The application, furthermore, provided an estimated schedule by years for subsequent receipts, consumption and transfer of SNM, consisting of needing no additional fuel until 1964, at which time it would need an additional 10 grams, and would not need an addition 10 grams again until 1969.[8]

For the next decade UCLA operated with less than 3.5 kg of U-235 total on site. After the first year of operation it discovered it did not need the Plutonium source for startup, as mentioned above, and received Amendment 2 to its license permitting it to use a radium source instead.

After a decade of operation with 3.5 kg, UCLA requested

6/ Construction permit application, 5/30/59, attached hereto.
7/ id., at page 7
8/ id.

"an additional 4.3 kg of Uranium-235 for the purpose of refueling." [9] This request, made June 3, 1970, indicated that refueling was to be done during the summer of 1971, and that therefore they would briefly have 7.8 kg on site, and "after refueling and shipment of the old fuel bundles approximately 4.3 kg." UCLA thus asked for permission to temporarily have on site 7.8 kg during refueling.

The AEC responded on June 24, 1970,[10] by reminding UCLA that its current possession limit was 4.0 kg, not 3.5 as UCLA thought (because for the previous decade that was all it had had on site), and thus asked UCLA whether it wanted the new license limit to be 7.8 or 8.3 kg. The University responded on July 9, saying "we would like the new limit to be 8.3 kilograms," and indicating that plans were for refueling in June of 1971.[11]

Two months later UCLA asked that the limit be altered further, this time to 10 kg, because the fuel manufacturer assertedly needed extra melt stock, scrap from which would be returned to UCLA.[12]

On October 26, 1970, just six weeks thereafter, the AEC published notice that it was amending UCLA's license from a limit of 4.0 kg to 10 kg, a 250% increase. The notice of issuance of facility license amendment 8[13] stated that the purpose for the amendment was that

> The additional material is required for the fabrication of fuel elements which will be used to replace those now in the reactor.

However, refueling never took place, the original fuel is still in the reactor, and the spare core loading and additional extra bundles and the scrap remained on site for a decade, unneeded and largely unprotected.[14]

9/ Letter, 3 June 1970, from Thomas Hicks, NEL Director, to Dr. Peter Morris, Division of Reactor Licensing, USAEC, attached.
10/ Letter, 24 June 1970, from Donald J. Skovholt, Assistant Director for Reactor Operations, Division of Reactor Licensing, USAEC, to Dr. Hicks, NEL
11/ Letter, 9 July 1970, from Dr. Hicks, NEL, to Donald Skovholt, USAEC
12/ Letter, 10 September 1970, from Dr. Hicks, NEL, to Dr. Morris, USAEC
13/ Notice of Issuance of Facility License Amendment, 26 October 1970
14/ The matter of the extraordinarily lax attitude towards protecting this sensitive material is the subject of Contention XX.

During the same period that UCLA was requesting amendments[15] to its license to increase U-235 possession limits to 10 kg, it was writing technical specifications as part of a relicensing request. Included in the latter request was a request for a 15-fold increase in its plutonium holdings, to 500 grams, and receipt in addition of 250 grams of U-233, as well as increase in authorized power level to 500 kw, or 50 times the level for which the reactor was initially designed. (Neither the plutonium nor the U-233, obviously, were to be used in the power increase. UCLA had already increased power to 100 kw a few years earlier.)

On August 31, 1970, the AEC, following a preliminary review of the above application, asked for the following additional information[16] to complete their evaluation:

1. A description of the physical form of the 250 grams of uranium-233 and the additional 467 grams of plutonium requested and a description of the proposed use of this material.

2. A supplemental safety analysis report, as described in Section 50.34 of 10 CFR Part 50, in support of your request to increase the power level to 500 kwt.

The University responded on October 7, withdrawing the request for the 250 grams of U-233, the 467 grams of Plutonium, and the increased power level.[17]

As will be indicated in CBG's brief on the issue of which set of security regulations (10 CFR 73.60 or 67) applies to this license request, having nearly 10 kg of highly enriched uranium on site during the 1970s became quite a compliance problem for UCLA and the Commission.

15/ Letter, February 20, 1970, from David Saxon, UCLA Vice Chancellor, to Donald Skovholt, Assistant Director for Reactor Operations, USAEC
16/ Letter, Skovholt to H.V. Brown, UCLA, August 31, 1970
17/ Letter, NEL Director Hicks to Dr. Peter Morris, USAEC, 7 October 1970

First, the AEC rejected UCLA's security plan because, even with the exemption for irradiated fuel, UCLA had a "formula"[18] quantity of SNM on site with a security plan, then as now, not sufficient to protect that quantity.[18a] UCLA promised to ship out just enough fuel to get it under the formula limit in order to avoid threatened enforcement action by the AEC.[19] A few years later an NRC inspection discovered UCLA still had more than a formula quantity; as NEL Director Catton put it, "We are presently in technical violation of our SNM possession limit, and further delay [in reducing inventory] could invite a Notice of Violation by the Nuclear Regulatory Commission."[20] The delay lasted an addition year and a half, with the shipment finally occurring in June of 1980, with disastrous results.[21] CBG contested UCLA's license renewal request, in part contending the SNM amounts were still excessive; an NRC site visit then confirmed this assertion, determining once again that UCLA had a formula quantity of SNM and had to take measures to better protect it or to reduce the inventory.[22] And just six weeks ago, in the midst of the Board hearing argument on whether UCLA had a formula quantity and therefore had to meet a higher standard of security protection for the HEU, the University notified the Board that it had shipped off just enough fuel to get below the 5.0 limit (i.e., UCLA claims to now have 4.92, though it appears to have forgotten about the Plutonium source, which according to the SNM formula, brings it once again over the formula level). There has been no confirmation of the actual amount currently on site; available records appear to contradict the 4.92 assertion, indicating the actual amount about half a kilogram higher. (Letter of October 28, 1972, from UCLA's Ashbaugh

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18/ a "formula" quantity of SNM is essentially that quantity defined by NRC as sufficient to make a bomb from without need of additional SNM. The formula quantity, as in 10 CFR 73.60, is grams SNM enriched over 20% + 2.5 (grams U-233 + grams Plutonium) = 5000 grams or more.

18a/ 18 November 1974 letter AEC George Lear to UCLA's Hicks

19/ 27 November 1974 letter, Charles Ashbaugh (UCLA) to AEC's Goller

20/ 1 March 1979 letter, UCLA's Catton, to DOE's Berger

21/ This was the shipment that took the wrong route, apparently so the driver could pick up his girlfriend and take her with him to Las Vegas, where the truck was parked overnight in a casino parking lot, and later found to be highly contaminated.

22/ January 29, 1981 Letter, NRC's Miller to UCLA's Wegst

to AEC's Goller, indicates that after shipping out the scrap UCLA was to have a total inventory of 9.047 kg. Records provided by UCLA's Neill Ostrander, transmitted by cover letter of William Cormier on August 26, 1982[23] indicate only two shipments took place thereafter, one of 730 grams and one of 2360 grams, which would leave the University still with about 5350 grams.)

In short, there are two periods in the history of the UCLA reactor's handling of HEU. For the reactor's first decade, the facility operated with less than 3.5 kg and was able to perform all its necessary functions. For the second decade, the University had on site nearly three times that amount, and was in constant violation of NRC security regulations because the amount on site was repeatedly found to be excessive. And further, the additional SNM never was used for the purpose requested, full core refueling, and the old core was thus never shipped off site, as promised, so what started out as a request to have 7.8 kg on site temporarily during refueling became an almost permanent period with nearly 10 kg, which was neither used nor needed. And now UCLA has shipped offsite some of the excess, but is refusing to amend its license or license application down to the level its security system can handle and the level it really has a need for.

B. UCLA Does Not Need a License for 9400 grams of U-235

The reactor can only operate on about 3.5 kg.[24] Burnup is extremely small, approximately one gram per year.[25] The maximum burnup that could take place in the requested license period is less than 40 grams total.[26] Even taking into account potentially clumsy fuel handlers who might

23/ in response to agreement between Applicant and CBG, Applicant clarified its interrogatory responses through a submission on August 26, 1982
24/ id
25/ see Hafemeister declaration; also Ostrander memorandum, cited above
26/ Hafemeister declaration

damage fuel during loading or unloading, a maximum of 700 grams, including burnup, can be expected to be needed during the next twenty years, based on the reactor's operating history.[27] Given the current core arrangement and maximum burnup, the facility needs only 3600 grams through the year 2000, the requested license period. Counting in clumsy fuel handlers brings a total of 4300 grams--on the assumption that the full twenty years' spare supply should be kept on site all the time rather than shipped on when needed, as was the original arrangement with the AEC when UCLA first was licensed.[28]

But the UCLA reactor can perform its intended function on even less than that. The Battelle study[29] cites the minimum critical mass for the Argonaut as 1.9 kg with one-slab geometry, and as 2.2 kg kg with slightly increased spacing of the fuel elements (the Argonaut is undermoderated, so increasing the spacing increases the volume of water between plates, and hence the moderation.) These figures are based on experience with different core configurations in the original Argonaut at Argonne National Labs, which used 20% enriched fuel.[30]

Thus it is indisputible that the UCLA Argonaut reactor can perform its licensed functions with a far smaller inventory of SNM. It is indisputible because for a decade UCLA did precisely that, because UCLA asserts it currently possesses about half of the amount of SNM it has requested in its license application, and with some relatively minor modifications to the geometry of the core, it can operate on roughly one fifth the requested amount. How far below 9400 grams UCLA should be required to go is perhaps disputable; there can be no dispute that 9400 grams is too much. That is close to enough for two atom bombs, if diverted or stolen. When the core only holds 3550 and burnup is a gram a year, 9400 grams is excessive and an unnecessary

27/ Environmental Impact Appraisal, page 5
28/ see Appendix A to original Facility License
29/ NUREG/CR-2079, page 23,
30/ Argonaut Reactor Databook by Sturm and Daavettila, ANL-6285, January 1961; Summary Report on the Hazards of the Argonaut Reactor, Lennox and Kelber, ANL-5647, December 1956

risk to public health and safety and the common defense and security.

VI. THE REQUESTED ENRICHMENT IS EXCESSIVE AND UNNECESSARY

A. History

In the 1950s and 1960s, low power research reactors were built in many countries, including the U.S., which utilized flat plate MTR-type fuel containing 20% or less enriched uranium, a value chosen because it was considered to be a limit for weapon usable material.[31] Highly Enriched Uranium (HEU) came into demand for high power research reactors, and eventually many low power reactors for which Low Enriched Uranium (LEU) would have sufficed were using HEU instead.[32] In the 1970s, however, particularly after India exploded a nuclear weapon using nuclear material obtained through a research reactor, concern grew once again about the use of HEU in research reactors (as well as very low enriched uranium in which Plutonium can be generated). This concern led to a national policy of attempting to reduce enrichments of research reactor fuels and reduce the amount of HEU in use.[33]

The Argonaut reactor has a similar history. The original Argonaut at Argonne National Labs used 20% enriched fuel [34] (in fact, the uranium was in oxide form, which has other useful safety and non-proliferation properties). The first Argonaut used 20% fuel for many years, from the time it first went critical, in February of 1957, through the time of most recent reporting, October of 1962[35] (the original Argonaut was dismantled and no longer exists).

31/ See IAEA-TECDOC-233, "Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels Guidebook", a Technical Document issues by the International Atomic Energy Agency, Vienna, 1980, p. 1, attached.

32/ id

33/ id

34/ Summary Report on the Hazards of the Argonaut Reactor, ANL-5647, by Lennox and Kelber, December 1956

35/ IAEA Directory of Nuclear Reactors, Volume V, 1964; also, Argonaut Reactor Databook, by Stumm and Pravettila, January 1961

The first commercially-available Argonaut in the U.S., built by AMF for the University of Florida, likewise used 20% enriched fuel.[36] In fact, the University of Florida continued to use 20% fuel until 1970, when it replaced its original core.[37] Now, because of the new policy of reduced enrichments for research reactors and the heightened concern about HEU, the University of Florida is involved with a program with DOE to use 4.8% enriched SPERT fuel.[38] The policy of reducing both the quantity of HEU in use and the enrichment of research reactors has been official U.S. policy since 1977[39]; the policy "has been fully supported by NRC since its inception.[40] NRC itself has issued a formal statement of policy declaring that in exercising its licensing responsibility for domestic use and export abroad of SNM, the NRC is interested in reducing, "to the maximum extent possible," the use of HEU in domestic and foreign research reactors.[41] One of the issues before the Atomic Safety and Licensing Board is how to reconcile UCLA's request for 9400 grams of 93% enriched HEU with the Commission's policy of reducing, "to the maximum extent possible," the use of HEU in domestic and foreign research reactors. As we shall see, that reconciliation is relatively easy: UCLA doesn't need HEU in order to perform its intended activities.

B. UCLA Doesn't Need HEU

There is no dispute about the material facts: Argonaut reactors like UCLA's can ran on LEU. They can because they do. In addition to the original Argonaut and the University of Florida Argonaut, the following Argonaut-type reactors are listed in the IAEA Directory of Research Reactors[42]

36/ University of Florida Training Reactor Hazards Summary Report, A Report to the USAEC from the Department of Nuclear Engineering at the University of Florida, 1958
37/ NRC Staff Answer to Interrogatory 89 as to the Safety Evaluation Report (3/17/82)
38/ Letter, October 10, 1978, to Robert Reid, USNRC, from N.J. Diaz, University of Florida
39/ See Hafemeister declaration
40/ USNRC Statement of Policy: "Use of High-Enriched Uranium in Research Reactors"
41/ id, 47 FR 37007, August 24, 1982

as having 20% enriched fuel: Siemens Argonaut Reactor No. 1 (Germany), Siemens Argonaut Reactor Karlsruhe (Germany), AEG Prüfreaktor PR-10 (Germany), Reattore Argonaut AGIP-NUCLEARE (Bologna), Siemens Argonaut Reactor Graz (Graz).

As indicated in the Hafemeister declaration and the Congressional testimony by DOE officials cited therein, LEU fuels are currently available, using available technologies and core designs, for low power research reactors such as UCLA's. It appears to be only the few, very high power research reactors who might need to await the commercialization of higher density fuels; even that seems available very shortly. DOE's annual reports [43] give targets of 1982 and 1984 for demonstration of the fuels being completed. IAEA has published a detailed handbook on how to make the conversion, and other assistance is available. General Atomics, for example, currently has available TRIGA low-enriched zirconium hydride fuel for use in converting and upgrading existing MTR plate-type reactors.[44] In addition to significantly reducing proliferation concerns, the TRIGA fuel would immeasurably add to the safety of this particular reactor at UCLA because of its marked ability to prevent reactivity accidents because of the instantaneous negative temperature coefficient.[45]

In response to CBG interrogatory XIII/11 of 4/20/81, UCLA stated that it knew of no reason why the reactor couldn't function if the enrichment level were reduced from 93% to 20%. In light of the overwhelming evidence that Argonauts can function on LEU, have and do function on it, and that LEU is available, and in the face of Applicant's knowing of no reason why the enrichment shouldn't be lowered to 20%, and given the NRC's policy in this regard, CBG is entitled as a matter of law to a favorable determination on its contention that the requested enrichment is excessive.

43/ DOE/NE-001, Nuclear Proliferation and Civilian Nuclear Power: Report of the Nonproliferation Alternative Systems Assessment Program, USDOE, June 1980

44/ IAEA Conversion Handbook, page B-2

45/ Because the moderator is part of the fuel, fuel heat-up immediately heats up the moderator, causing power to drop and preventing destructive excursions

VII. Legal Argument

In connection with its Part 50 Application to be licensed to operate its nuclear reactor, UCLA has requested a license for SNM. In order to issue a license for the possession of SNM, the Licensing Board must determine that the application meets the requirements of the regulations and that such issuance would not be inimical to the common defense and would not constitute an unreasonable risk to the health and safety of the public. 10 CFR 70.31.

Risk is defined as probability times consequences. The consequences of theft of 9400 grams of 93% enriched uranium can be immense. As the Commission put it while proposing stricter safeguards for SNM, the consequences of the successful detonation of a clandestine weapon would be "disastrous".[46] Thus, anything which would tend to increase the probability of theft or diversion of HEU, even by a small fraction, would vastly increase risk. For this reason, the Commission states in its recent Statement of Policy on the subject[47]:

> In an effort to allay concerns of proliferation risks, efforts were made to reduce HEU inventories, on the assumption that any reduction in the potential for access to these inventories would constitute a reduction in the proliferation risk.

The Commission argues further:

> The widespread use of HEU fuel, which involved a large number of domestic and international fuel shipments, increases the risks of proliferation through theft or diversion of this material. In contrast to HEU, the use of fuel with lower enrichments reduces proliferation risks.[48]

The Statement of Policy makes clear that it is the Commission's policy to both reduce the amount of HEU available through its licensing responsibility and the enrichment of SNM permitted.

46/ Proposed Rulemaking, Physical Protection of Plants and Materials, 43 FR 35323
47/ Statement of Policy: Use of High-Enriched Uranium in Research Reactors; 47FR37007
48/ UCLA's HEU shipment parked overnight in a casino parking lot underscores this poin

10 CFR 70.31 prohibits a Licensing Board from granting a requested license for SNM if the Board cannot determine that issuance of the license will not be inimical to common defense and security. In addition, the Board must determine that grant of the license would not result in unreasonable risk to public health and safety. As has been demonstrated above, the Commission is on record as recognizing that HEU increases the risks and LEU lowers risks; and that reducing the amount of HEU available for its theft or diversion reduces the risks associated thereto.

The risks associated with a license request for HEU are not unreasonable, in 10 CFR 70.31 terms, if there are no alternatives to its use in the amounts and enrichments requested and if the benefits outweigh the risks.

The Applicant in this case clearly does not meet that standard, because the requested license is in excess of need. Therefore the request poses an unreasonable risk and must, as a matter of law, be denied.

(1) The Request for 32 Grams of Plutonium Must be Denied. UCLA used the Plutonium-Beryllium start-up source for its intended use for only a year before requesting the Commission amend its license to permit use of a less hazardous Radium source and hasn't used the Pu-Be source for reactor operations in the twenty years since. Applicant thus doesn't need the requested Plutonium, and therefore grant of the license would pose an unreasonable risk and must be denied.

(2) The Request for 9400 grams of HEU Must be Denied. The core can only hold 3600 grams; the facility operated without difficulty for ten years with 3500 grams; the facility operated with difficulty in terms of compliance with Commission safeguards regulations during the subsequent ten years

while it had nearly ten kilograms; the Applicant is unwilling--and due to the technical difficulties involved with keeping the fuel at self-protecting levels discussed in CBG's 73.60 brief--incapable of safeguarding the amount of material requested; the reactor can run on about two kilograms; it burns up only 1 gram per year; and simply has no need nor even use for 9400 grams.

(3) The Request for 93% Enriched Uranium Must be Denied.

The material requested is weapons grade and in excess of the quantity necessary for constructing a clandestine fission explosive; it is NRC policy to reduce enrichments, including through its licensing authority; the reactor can (and other Argonauts have) run on LEU; the risks associated with grant of the requested license are greater if HEU is granted than if LEU is granted; the consequences are so disastrous that any small increase in risk is unreasonable unless no alternatives exist and benefits outweigh the risks; alternatives do exist and the benefits in no way outweigh the risks; therefore 93% HEU is not needed, poses an unreasonable risk, and must be denied.

There is another requirement that an applicant must meet before a Board can be permitted to issue the requested license: that is, the application must meet the regulatory requirements. In this case, UCLA has requested a license for nearly 10 kg of SNM but failed to provide the information required by the regulations (10 CFR 70.22 and .2) with regards how Applicant will protect against criticality accidents. The law prohibits grant of the application if the application violates the provisions of those regulations. The reason is simple: a licensing board cannot

possibly determine that grant of the proposed license will not pose an unreasonable risk to public health and safety if the Applicant refuses to provide the required information for Board review as to how Applicant intends to protect public health and safety should the license issue. Criticality accidents are no trivial matter; ask the families of Louis Slotin, the Woods River Junction victim, and the scores of others who have been seriously injured in the thirty or so accidents to date. But the bottom line is that the law prohibits grant of the license without the required information.

Lastly, summary disposition should be granted as a matter of law because no genuine dispute exists as to the material facts and the facts are such as to require such a ruling.

The University, by its recent reduction in SNM inventory, has admitted that the 9.4 kg it has requested is both excessive and unnecessary. By its arguments against the applicability of 10 CFR 73.60 and its repeated violations of safeguards regulations applicable to the amounts possessed during the last decade, the Applicant has indicated it is both unable and unwilling to protect the quantity of material requested. The license therefore cannot issue.

The Staff, in formal submission to the Commissioners, has committed itself to amending UCLA's license to reduce possession authorization below a formula quantity. Memo, SECY-81-376, "PHYSICAL SECURITY REQUIREMENTS FOR NONPOWER REACTOR LICENSEES POSSESSING A FORMULA QUANTITY OF SSNM", from William J. Dircks, Executive Director for Operations, to the Commissioners, dated June 12, 1981 states as follows:

> In SECY 79-187B, 22 nonpower reactor licensees were listed as having licenses to possess a formula quantity or more of SSNM. Of these 22, seven have taken or are taking action to reduce their holdings to less than a formula quantity of SSNM and the NRC will take action to amend their licenses to reduce possession authorization below a formula quantity.

UCLA is listed as one of the seven, confirming what CBG has alleged all along, that UCLA is licensed to possess more than a formula quantity of SSNM and should reduce its holding. And the above memo commits the NRC to taking the action recommended by CBG's contention: reducing possession authorization to at least below a formula quantity.

CBG believes the reduction should be below 4.92 kg, as UCLA claims, perhaps erroneously, it now possesses (but is unwilling to be licensed for). As the Commission stated in amending Part 73: "it can be properly argued that a 4.9 formula kilogram quantity of SNM is about as important a quantity as 5.0 kilograms." 44 FR 43281, July 24, 1979. And as Dr. Hafemeister points out in his declaration, 4.9 kilograms of 93% enriched uranium is nearly three times more dangerous from a nuclear proliferation standpoint that 5.0 kilograms of 20% enriched, because the critical mass for the 93% is one third as much as for 20% (i.e., one needs only a third as much 93% uranium to make a bomb).

VII. CONCLUSION

CBG has demonstrated through indisputible evidence that the amount and enrichment of SNM requested are excessive and thus pose an unreasonable risk as prohibited by 10 CFR 70.31. CBG has further demonstrated that there is no dispute that certain information required by the regulations to be provided in an application before approval can be granted is not in fact included. As a matter of law, CBG is entitled

to a decision in its favor on Contention XIII. No more important matter will be before this Board than the prevention of an unnecessary increase in the probability that a clandestine fission explosive might be acquired and detonated. The consequences of such an occurrence would indeed be, as the Commission has said, "disastrous." This Board can reduce those risks, and the law requires that it do so.

Respectfully submitted,

Daniel Hirsch

Daniel Hirsch
President
COMMITTEE TO BRIDGE THE GAP

STATEMENT OF MATERIAL FACTS AS TO WHICH NO GENUINE DISPUTE EXISTS

1. The full information as to equipment and procedures designed to prevent, mitigate, detect, and respond to criticality accidents required by 10 CFR 70.22(a)(7) and (a)(8) and 70.24(a)(1),(2), and (3) has not been provided in the Application.

2. The UCLA reactor can operate with less than 9.4 kg U-235 on site.

3. The UCLA reactor operated with approximately 3.5 kg of U-235 on site for ten years.

4. The UCLA reactor did not use more than 4.3 kg of U-235 during its entire lifetime to date.

5. The UCLA reactor burns up on the average approximately 1 gram of U-235 per year.

6. Total fuel damaged or burnt-up in the last twenty-two years is less than 750 grams.

7. 93% enriched uranium is weapons-grade uranium.

8. 93% enriched uranium is Highly Enriched Uranium (HEU).

9. Low Enriched Uranium (LEU) that is not so low enriched as to produce sizeable plutonium generation reduces prolieration risks.

10. It is official U.S. policy to reduce the enrichment of research reactor fuels.

11. Reduced enrichment fuels are currently available on which the UCLA reactor can run.

12. The original Argonaut reactor ran on 20% fuel.

13. The University of Florida reactor ran until 1970 on 20% fuel.

14. Other Argonaut reactors have operated on 20% fuel.

15. Advanced reduced enrichment fuels of higher Uranium loading will soon be available on which all but the highest power research reactors can run.

16. The UCLA reactor does not use a Plutonium-Beryllium neutron startup source.

17. The UCLA reactor uses a Radium-Beryllium startup source.

18. Two curies of Plutonium-239 are more hazardous if released to the environment than 6.6 millicuries of Radium.

19. UCLA does not have a security plan and safeguards contingency plan that meets the requirements of 10 CFR 73.60 for formula quantities of SNM.

20. The amount of SNM requested in the license, if all were on site, would be a formula quantity of SNM.

21. UCLA has recently reduced its SNM inventory.

22. The NRC has committed itself to reducing UCLA's license authority to below a formula quantity.

Exhibit A

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	
THE REGENTS OF THE UNIVERSITY OF CALIFORNIA	Docket No. 50-142 OL
(UCLA Research Reactor)	(Proposed Renewal of Facility License)

DECLARATION OF DAVID W. HAFEMEISTER

I, David W. Hafemeister, declare as follows:

1. I am presently Professor of Physics at the California Polytechnic University in San Luis Obispo, California. My professional qualifications are attached.

2. During the period 1975-1979, I was intimately involved with the development and administration of this nation's nuclear non-proliferation policy, both in the U.S. Senate and in the U.S. State Department. This work included domestic and foreign policy matters related to the use of High Enriched Uranium (HEU) in research reactors and methods to reduce the associated proliferation risks.

3. The proliferation risk associated with HEU is that it can be used directly to make nuclear weapons, unlike the low enriched uranium used, for example, in power reactors. No further enrichment, generally very costly and difficult, would be necessary in order to utilize the material in a clandestine fission explosive, thus making it a potentially attractive target for theft or diversion. For this and related reasons, it has been the policy, both nationally and internationally, to attempt to minimize the amount of HEU in use.

4. 93% enriched uranium in flat plate Aluminum-Uranium fuel would clearly fit within the category of Highly Enriched Uranium. In fact, 93% would be near the upper limit of HEU normally used in reactors, and is clearly "weapons-grade." That is, it could be used directly to fashion a clandestine fission explosive. Furthermore, because the critical mass goes down as enrichment goes up, one would need significantly less U-235 if 93% enriched than, say, 20%, for which the critical mass of U-235 is roughly three times as large. Thus, 93% enriched uranium poses significant proliferation risks and requires significant safeguards if its use is essential.

5. The prevention of nuclear proliferation is a matter which has long been recognized as essential to U.S. interests and the common defense and security. The solutions to nuclear nonproliferation are not simple: The office of Technology Assessment report on Nuclear Proliferation (1977) says that:

> It is not too late to contain proliferation at a level which can be assimilated by the international political system. However, there are no single or all-purpose solutions; no short-cuts. A viable nonproliferation policy will require the coordinated, planned use of a wide variety of measures...

6. **In recognition of the threat to** common defense and security posed by nuclear weapons proliferation, the Congress passed (virtually unanimously) the Nuclear Nonproliferation Act of 1978. And, beginning in 1977, the United States Government established a policy designed to reduce the threat of **proliferation** by attempting to reduce the risk of theft or diversion of HEU, in part by attempting to reduce the amount of HEU in use throughout the world, particularly for research reactors. This policy of reducing the threat of theft or diversion by reducing the amount of HEU available for theft or diversion has had as a concommitant element the attempt to reduce the enrichment of research reactor fuels. This program, known as the Reduced-Enrichment Research and Test Reactor Program (RERTR), represents the official policy of the United States in attempting to reduce enrichments of research reactor fuels and thus the amount of HEU in use.

7. The summary report of the International Nuclear Fuel Cycle Evaluation (1980) has stated that it is feasible to markedly reduce the uranium enrichment of a great majority of research reactors; INFCE endorsed the conversion of HEU fueled research reactors to lower enrichment. As C. Worthington Bateman, Acting Under Secretary of Energy in 1980, testified to the Congress that with fuel fabrication technology presently available in the U.S. and Europe enrichment reduction is possible for a great many reactors. And John N. Deutch, then-Director of Energy Research at DOE, told Congress in 1979 that fuel fabrication and core technology currently available in the U.S. and Europe permits enrichment reduction from 90-93 percent to below 20 percent in most reactors. Mr. Bateman indicated in his testimony that the easiest reactors to make use of reduced enrichment fuels are low power reactors. The Department of Energy's NASAP Program stated in 1980 that for those reactors where conversion using current technology might be difficult, substitution of higher uranium density fuels with lower enrichment should be possible. In this way the density (g/cc) of U-235 would remain essentially a constant, but the additional U-238 atoms would dilute the U-235 so that it would be less useable as a material for nuclear weapons.

8. Given the official U.S. policy of reducing the amount of HEU in use to that absolutely essential, and the policy of reducing research reactor fuel enrichments, it is my opinion that UCLA's request for a license for 93% enriched fuel should not be granted unless the applicant can show definitely that it cannot adequately operate the reactor without HEU of that enrichment.

9. Likewise, UCLA's request for a license to possess, as I understand it, 9400 grams of U-235 at 93% enrichment seems to me to necessitate a very substantial showing on the Applicant's part why such a very large amount of such sensitive material could ever be needed on site. If it is true that the core loading is about 3600 grams, it seems to me an unnecessary risk for the facility to be permitted to have on site much more than a few hundred grams beyond that. Burnup would appear to be minimal. The rule of thumb is that 1 gram of fissionable material is burned up per MWD of heat produced; given a maximum power level of 100 kW_{th} and a restriction to 5% of the year operating factor, which I am told the reactor is restricted to, in 20 years a maximum of about 36 MWD of thermal energy could be produced. If this is so, a maximum of less than 40 grams of U-235 will be consumed through burn-up, a far cry from the thousands of grams requested in the license.

10. I understand that the Environmental Impact Appraisal for this reactor indicates that a total of about 700 grams of U-235 have been "used" in the past twenty years. If this is true, and assuming that part of that 700 grams constitutes damaged fuel as opposed to burnup, operating experience would indicate approximately 700 grams spare fuel would be sufficient, and even then, there is no reason of which I am aware that a full twenty years' supply needs to be on site all the time or at any one time. In my opinion, more than 4300-4500 grams U-235 permitted on site and granted through a license would be excessive, absent a substantial showing of need, and would pose an unnecessary threat to common defense and security through risk of diversion or theft.

11. I have reviewed a July 1982 calculation by Neill C. Ostrander of the Nuclear Energy Laboratory entitled "Fuel Self Protection Calculation." If he is correct that after seven days of shutdown the radiation dose at four feet from the core center without intervening shielding is 142 Rem/hour, then each individual fuel bundle (of which I understand there are twenty-four, each containing eleven fuel plates) would be about 10 Rem/hour at three feet (unshielded). Thus it would appear necessary to raise these radiation levels by more frequent (short-term) operation of the reactor to approach the 100 Rem/hour level for each fuel bundle and would appear prudent to do so if the radiation level of the fuel is being relied upon as a deterrent to theft.

12. My conclusions are that the Applicant, in order to obtain a license, should: (a) reduce the total amount of U-235 permitted on site to about 4 kg, (b) lower the enrichment of U-235 significantly unless the Applicant can clear demonstrate that this is infeasible, and (c) institute an operation schedule which would raise the radiation level of the fuel bundles. In addition, the security measures taken to protect what material is permitted on site need to be substantial, particularly if the above measures are not taken. 9700 grams of 93% enriched uranium are by no means de minimus; nor for that matter are 4900 grams. Theft or diversion of such material could have grave effects for our common defense and security, as well as public health and safety.

13. The above suggestions would be consistent with U.S. policy and prudent in terms of protecting against the very worrisome prospect of an unnecessarily large quantity and unnecessarily high enrichment of uranium without adequate safeguards being stolen or diverted for use in a clandestine fission explosive. Furthermore, however, failure to take the above precautions, without substantial showing of good cause not to, would damage U.S. foreign policy interests by undercutting our government's attempts to reduce international commerce in HEU and convince other nations of the need to reduce their HEU holdings and the enrichment of their research reactor fuels. I know from personal experience in representing the State Department in such interactions with Chilean nuclear officials and representatives of Atomic Energy Commissions of other nations that it will be much more difficult for the U.S. to succeed in its policy of reduced enrichments and HEU holdings abroad if the policy is not vigorously pursued at home. The inconsistency of the US, on the one hand, denying HEU to foreign research reactors while, at the same time, oversupplying research reactors at home with HEU that is not properly safeguarded, would not be lost on the nations we are trying to influence.

14. Lastly, it should be stated that it is both national and international policy that kilogram quantities of HEU must be safeguarded. While timely warning, after the fact, of theft or diversion is a key element in such safeguards, post-loss reporting is not sufficient protection and, in my opinion, fails to meet the standard of taking measures to minimize the possibilities for unauthorized removal of such material consistent with the consequences of such removal. The removal of 9400 grams of 93% enriched U-235 would have extraordinarily serious potential consequences; the removal of 4900 grams of such material would have potential consequences many, many times greater than removal of 1000 grams of 20% enriched uranium. But even 1000 grams of such material, given the world situation with regards pressures for nuclear weapons proliferation, is not de minimus.

I, David W. Hafemeister, swear under penalty of perjury under the laws of the U.S. that the foregoing is true and correct to the best of my knowledge and belief.

Executed on August 25, 1982,
at Santa Cruz, California

David W. Hafemeister

David W. Hafemeister, Ph.D.

David W. Hafemeister
Professor of Physics
California State Polytechnic University

Professional Qualifications

1. Education

   a. Bachelor of Science degree in Mechanical Engineering from Northwestern University, 1957
   b. MS. and Ph.D. in Physics, University of Illinois, 1959, 1964
   c. Post-Doctoral Fellowships:
      Los Alamos Scientific Laboratory (1964-66)
      American Association for the Advancement of Science
      Congressional Fellowship (1975-1976)

2. Employment

   a. Mechanical Engineer, Argonne National Lab (1957-58)
   b. Physicist, Los Alamos Scientific Laboratory (1964-66)
   c. Assistant Professor of Physics, Carnegie-Mellow University (1966-69)
   d. Associate Professor of Physics (1969-72)
      Professor of Physics (1972-)
      California Polytechnic University, San Luis Obispo, CA
   e. Visiting Professor of Physics
      University of Groningen, The Netherlands (1972, 1980)
   f. Legislative Assistant and Science Advisor to Senator John Glenn
      U.S. Senate (1975-77)
   g. Special Assistant to Under Secretary of State Lucy Benson and
      Deputy-Under Secretary Joseph Nye, U.S. Department of State (1977-1979)

3. Experience with Nuclear Non-Proliferation Matters

   a. U.S. Senate: After the detonation by India of a nuclear device in 1974, the Committee on Governmental Affairs of the U.S. Senate held extensive hearings on the "Export Reorganization Act of 1975" which dealt with nuclear nonproliferation. It was my job to be the full-time staffperson to the Ad-hoc Chairman of the Committee, Senator Glenn, on hearings and mark-up of the act. I was Senator Glenn's main advisor on nuclear non-proliferation matters.

   b. Department of State: In 1977, I was appointed as one of two Special Assistant on the issue of nuclear nonproliferation to Under Secretary Benson and Deputy-Under Secretary Nye. Dr. Nye had the lead role for nuclear non-proliferation in the Executive Branch and at the London Nuclear Supplier Negotiations.

      During this time, I was intimately involved with the drafting and passage of the Nuclear Non-Proliferation Act of 1978, participating in the Department of Energy's Non-proliferation Alternative Systems Assessment Program (NASAP), and dealing as a representative of the Under Secretary with officials of other nations' nuclear programs.

In addition, I was the lead State Department delegate to Working Group 8 (Advanced Fuel Cycle and Reactor Concepts) of the International Fuel Cycle Evaluation (INFCE) which was held at the International Atomic Energy Agency (IAEA) in Vienna. Subgroup C of this Working Group had as its sole task the assessment of methods of reducing proliferation risks associated with research reactors.

4. Publications

a. Nuclear Non-Proliferation:

i. "Nonproliferation and Alternative Nuclear Technologies", Technology Review 81, 58 (December 1978).

ii. "Science and Society Test V: Nuclear Nonproliferation", American Journal of Physics 48, 112 (1980)

iii. prime author/editor of the Presidential Report to the Congress on the environmental impacts associated with nuclear exports abroad (1980)

iv. co-author/editor of the Supplement Nuclear Research and Development Export Activities to ERDA 1542 (U.S. Nuclear Export Activities), September 1979.

b. Solid State and Nuclear Physics:

20 articles; four book chapters; one book

c. Energy Technology and Policy:

10 articles

CONGRESS OF THE UNITED STATES
Office of Technology Assessment

Exhibit B

Nuclear Proliferation and Safeguards

The Praeger Special Studies program—utilizing the most modern and efficient book production techniques and a selective worldwide distribution network—makes available to the academic, government, and business communities significant, timely research in U.S. and international eco-

Praeger Publishers New York London

political dissidents. A second position treats safeguards as an acceptable extension of existing clearance programs and blackmail threat responses in other fields of high security. A third position believes safeguards could be installed without doing serious damage to civil liberties, but only if a "least intrusive measures" approach is adopted and a zero-risk goal is rejected.

Although a safeguards system that would be extremely respectful of civil liberties can be designed, three potential dangers exist:

1. A gradual erosion of civil liberties as the safeguards system is "strengthened,"
2. A shunting aside of civil liberties during a recovery operation if weapons material were diverted and a convincing threat received; and
3. A public demand for Draconian safeguards in the future, even at the expense of civil liberties, if a diversion followed by a convincing threat or an actual act of destruction occurred.

Measures can be envisaged that would reduce the probability of the above three occurrences. Continued public monitoring of safeguards systems for civil liberties infractions, new technologies or configurations (e.g., coprecipitation or colocation), and response planning integrated at the local, State, regional, and Federal levels with authority clearly delineated could reduce the probability of civil liberties infractions in a strong safeguards system.

The Control

Issue 15

What is the Outlook for Control of Proliferation?

Findings

It is not too late to contain proliferation at a level which can be assimilated by the international political system. However, there are no single or all-purpose solutions; no short cuts. A viable nonproliferation policy will require the coordinated, planned use of a wide variety of measures: (a) political, economic, institutional, technological; (b) unilateral, bilateral, multilateral, international; and (c) executive and legislative.

Components of a nonproliferation policy would include: (a) Steps designed to tip the balance of political incentives and disincentives regarding the acquisition of weapons in favor of disincentives; (b) A comprehensive safeguards regime to prevent the diversion of nuclear material from civilian energy programs to weapons use; (c) Controls over exports, particularly with regard to enrichment and reprocessing capabilities, in conjunction with arrangements for the return of spent fuel to the supplier or any international repository; (d) A broad range of domestic and foreign policy supporting actions, including steps to upgrade physical security measures to prevent theft of nuclear materials, expansion of reactor-grade uranium production to obviate the need for reprocessing, and arms control negotiations; and (e) Steps to assure that other countries can meet their energy requirements without resorting to enrichment and/or reprocessing national facilities.

Moreover, because each Nth country is to some degree unique, policy must be tailored to fit particular national circumstances. This is especially true because of the potential for serious conflict between nonproliferation and other foreign policy objectives. The nature and severity of that conflict will vary from one Nth country to another, a fact which policy must take carefully into account. (Chapters III and IV.)

Issue 16

What Influence Can the United States Exert Upon Potential Weapons States?

Findings

In the long run two general rules apply: (a) Solutions to the proliferation problem will have to be found primarily, though not exclusively, through multilateral actions, and (b) The extent of U.S. influence will vary from country to country.

As American preeminence in the international market for nuclear fuel, facilities, and technology has been allowed to erode, the ability of the United States to unilaterally determine the ground rules of international nuclear cooperation has diminished. With the entrance of other suppliers into the market, importers have the option to turn to non-U.S. sources. If the United States were to remove itself from the global market entirely, other suppliers could quickly replace the withdrawn capacity. As a consequence American actions will tend to be most effective in a multilateral context—particularly in conjunction with other suppliers. The effectiveness of this approach has been demonstrated in the negotiations which led to the NPT, and more recently in the Suppliers' Conference.

There remains, however, significant scope for the unilateral assertion of U.S. influence—both in terms of positive inducements and negative sanctions. The recent successful U.S. effort inducing South Korea to abandon plans for purchasing a French reprocessing facility is an instance of the effective use of unilateral influence. Some of the more obvious levers available to Washington include:

- security guarantees;
- assistance to civilian nuclear energy programs;
- foreign economic aid (including U.S. influence in international lending institutions);
- military assistance programs;
- political pressures and diplomatic persuasion;
- mediation of international disputes with proliferation implications;
- controls on the export of sensitive nuclear technology;
- assistance concerning non-nuclear energy sources; and
- domestic policy initiatives (e.g., concerning reprocessing) which might enhance the credibility of U.S. efforts to persuade other countries to take similar steps.

The single most effective instrument of U.S. influence would be the capability to guarantee adequate low-enriched uranium exports to meet the needs of overseas users while, at the same time, providing for the collection a return of spent fuel.

An effective effort to assert U.S. influen will combine the carrot and the stick, w principal reliance on the former for the long term. Such an effort will also take into accou the wide variation in leverage available Washington when dealing with one N country or another. Thus U.S. influence wi nations dependent upon American military economic assistance (e.g., South Korea) is ve substantial but where such dependence lacking (e.g., Argentina) U.S. influen declines.

Issue 17

What Influence Can the United States Exert Upon Other Supplier States?

Findings

Efforts by the United States inducing oth supplier states to pursue policies supportive nonproliferation will generally be most effe tive if they are formulated in a multilater context and emphasize positive inducement Possible measures include:

- political-diplomatic persuasion (e.g., th Suppliers' Conference),
- tie-in agreements guaranteeing U.S enrichment services at nondiscrimina tory prices to reactor customers of othe suppliers,
- joint-venture enrichment and/o reprocessing facilities,
- market sharing agreements,
- multinational enrichment and/o reprocessing facilities,
- international fuel storage repositories and
- a multilateral study of alternatives t reprocessing.

The problem of reprocessing is extremely difficult for two reasons. First, other supplier states (such as Germany) have already made a basic national decision in favor of reprocessing and the breeder. They regard this policy as a vital element in their efforts to assure ade quate energy in the future. European breeder

EXHIBIT C

INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION

ADVANCED FUEL CYCLE AND REACTOR CONCEPTS

REPORT OF
INFCE WORKING GROUP 8

PUBLISHED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1980

CHAPTER 4

RESEARCH REACTORS: SUBGROUP 8C

4.1. INTRODUCTION

Subgroup C of Working Group 8 of INFCE is concerned with research reactors. The objective of this Subgroup is to review and report on the contributions[1] on this subject submitted by the various participating countries and organizations.

The term 'research reactor' is used here for thermal-neutron reactors that are designed, built and used as neutron and gamma-ray sources for fundamental research material irradiations, isotope production, fuel element and reactor safety tests, training etc. (Appendix 1). Over 150 research reactors of significant power (between 10 kW and 250 MW) are in operation with highly enriched uranium in more than 35 countries with a total power in excess of 1700 MW. The overall annual fuel requirement of these reactors is more than 1200 kg of ^{235}U.

The number of operating research reactors in the world does not appear to be increasing because the construction of new reactors is being offset by the decommissioning of older reactors. To satisfy cost-benefit considerations, new reactors are built only on well defined requirements and when excess capacity of existing reactors cannot be used effectively.

The proliferation aspect of the widely distributed highly enriched uranium and of the production of fissile materials in research reactors make these reactors of concern to INFCE. Subgroup C has, on the basis of a limited number of contributed studies, considered steps that might be taken to reduce proliferation risks without jeopardizing the function of research reactors.

In this report, the non-proliferation considerations are discussed in general terms in section 4.2. In particular, studies of possible enrichment reduction in research reactors to improve proliferation resistance are presented in section 4.3, which includes discussions on general criteria and considerations (section 4.3.1). Since the enrichment reduction potential depends on the fuel technology, this is also discussed (section 4.3.2). A short review of individual case studies is given in section 4.3.3, and their summaries as contributed to INFCE are included in Appendix 2. These summaries are the responsibility of the individual contributors and therefore, do not represent consensus of Working Group 8. Section 4.4 briefly discusses the research and development requirements associated with enrichment reduction. Section 4.5 presents the special needs of developing countries.

[1] Reference numbers are shown in brackets and are listed at the end of the chapter.

4.2. NON-PROLIFERATION CONSIDERATIONS

To maximize neutron flux per unit power and/or to minimize capital and fuel cycle costs many research and test reactors were designed or converted to utilize uranium enriched up to more than 90% in ^{235}U. On the other hand, a number of research reactors have also been designed for operation with very low enriched or natural uranium fuel.

Concerns over the use of highly enriched uranium in research reactors arise from the fact that feedstock materials, fresh and spent fuels containing highly enriched uranium represent a potential source of weapons-usable materials. A decrease to below 20% enrichment is internationally recognized to be a fully adequate isotopic barrier to weapons usability [2]. Therefore, although it may not be technically possible in some research reactors, decreasing the enrichment from the 90% range as far as reasonable toward 20% would be a worthwhile improvement in proliferation resistance of research reactor fuels.

The plutonium in spent fuels is also of concern, although attainment of weapons-usable material would require spent fuel reprocessing. The annual plutonium production is roughly proportional to the power level and to the amount of ^{238}U in the reactor and therefore decreases with increasing enrichment. Decreasing the annual plutonium production would have non-proliferation benefits. However the use of research reactors for fissile materials production is not prevented by changing fuel enrichments and, therefore, appropriate safeguarding of the reactor is still required.

In an overall assessment of the proliferation risks of a particular research reactor, it is necessary to consider both the enriched uranium as well as the plutonium produced, and adequate safeguards must be provided. Note that fissile materials are much less accessible in spent fuel because of their high radiation levels.

EXHIBIT D

DEPARTMENT OF ENERGY FISCAL YEAR 1981 BUDGET: NUCLEAR NONPROLIFERATION PROGRAMS

HEARING

BEFORE THE

SUBCOMMITTEES ON INTERNATIONAL SECURITY AND SCIENTIFIC AFFAIRS

AND ON

INTERNATIONAL ECONOMIC POLICY AND TRADE

OF THE

COMMITTEE ON FOREIGN AFFAIRS HOUSE OF REPRESENTATIVES

NINETY-SIXTH CONGRESS

SECOND SESSION

APRIL 16, 1980

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WITNESSES

MATERIAL FOR THE RECORD

The Office of Nonproliferation under the Director of Energy Research plays an important policy and technical coordinating role across a wide range of nonproliferation programs and issues.

The Office of Defense programs administers the safeguards and security program and export controls.

These offices have all given crucial support to DOE's involvement in our nonproliferation efforts. Their continued support will obviously be necessary for DOE to achieve its nonproliferation objectives.

You asked what changes have been made since Under Secretary Deutch left the Government. Basically the assignment of responsibilities that I have just described continues with the arrangement that prevailed during Dr. Deutch's tenure.

In addition we have assigned to the Assistant Secretary for Nuclear Energy responsibility for coordinating the flow of paper and departmental positions, responses, and reactions on specific issues. That work is coordinated through my office and more generally through the Office of the Secretary.

This arrangement reflects more realistically the ability of the different staffs to carry out responsibilities in this area.

Dr. Deutch assumed a leadership role in this area through many different jobs that he held in the Department of Energy. He himself recognized, prior to his departure, that his direct and continuing involvement was to some extent out of sync with his responsibilities as Under Secretary. Dr. Deutch moved immediately prior to his departure, to change that arrangement in a way which I think is a workable one and which he felt more accurately reflected the degree that the Under Secretary could be involved in these matters.

I think it is clear that there is no intent to diminish the coordination process, and certainly it is not intended to reflect any diminution of the role of the Under Secretary in these matters.

It simply reflects the range of activities that the Under Secretary is responsible for and the need to have day to day staff support assigned and delegated elsewhere.

With that let me say a few words about the specific programs which the Department of Energy is responsible for that support U.S. nonproliferation efforts.

First, I would like to mention the reduced enrichment research and test reactor program. The objective of this program is to develop and demonstrate technology that can use low enriched uranium fuels in research reactors now using HEU fuel. This is a move which has been supported by INFCE and NASAP. Foreign acceptance of this program is also widespread. Japan, France and the FRG all have R. & D. programs in this area. This is something we are committed to.

The current budget proposal is for $3 million which is somewhat lower than we may have liked but in the fact of very severe fiscal constraints and cutbacks we think this is a program which is viable. It emphasizes short-term and near-term results and basically puts us in a stretch-out mode with respect to some of the more advanced fuel development activities.

The LWR improvement program is also a very important one to us. It is clear we can realize very significant economies in uranium resource consumption by backfitting existing reactors. We are committed to carrying through this program. There is tremendous interest on the part of utilities and fuel vendors in cooperating with us in moving this forward.

We are proposing a budget of about $21 million in this area for fiscal year 1981.

The international energy development program was initiated in 1977 by President Carter. Two assessments have been completed in Egypt and Peru. The reception these studies have received is encouraging. We currently have studies underway in three other countries; Argentina, South Korea, and Portugal. We have high expectations for a similar response in those areas.

In the safeguards and security program, we are proposing a fiscal year 1981 budget of $50.4 million. Most of this money is for domestic safeguards and security but a significant part of the program does support international safeguard activities.

We are supporting R. & D. for physical protection hardware under bilateral agreements. Systems are being developed for possible application of safeguards for facilities such as enrichment plants and spent fuel storage. International training courses are given as required by the NNPA. Additional direct technical support to the IAEA is being provided through programs funded by the Department of State under the Foreign Assistance Act but managed by the DOE Office of Safeguards and Security.

The NURE program, which is administered by the Assistant Secretary for Resource Applications is continuing. The data collection, evaluation and assessment of U.S. uranium resources effort is being reduced in light of lower projections of nuclear capacity and nuclear demand.

We expect to complete the NURE program work by 1990. In fiscal year 1981 we propose a budget of approximately $30 million.

Mr. Chairman, that summarizes some of the high points of my testimony. I will not try to describe each program in the same detail. I would be happy to try to answer any questions you may have.

Mr. BINGHAM. Thank you, Dr. Bateman.

[Mr. Bateman's prepared statement follows:]

PREPARED STATEMENT OF C. WORTHINGTON BATEMAN, ACTING UNDER SECRETARY OF ENERGY

I am pleased to appear before you today to discuss the Department of Energy (DOE) programs and activities in support of U.S. nonproliferation policy. The Department, in close cooperation with the Department of State (DOS) and the Arms Control and Disarmament Agency (ACDA), has played a major role in U.S. nonproliferation efforts, and continues to attach a high priority to the development of nuclear fuel cycle approaches, both domestically and internationally, that minimize the risks that civilian nuclear power systems and research reactors might contribute to the spread of nuclear weapons.

We are at an important juncture in the nonproliferation efforts initiated by the Carter Administration three years ago. The extensive studies performed by INFCE and NASAP have been completed, although the full implications of their

findings must still be assessed. In the months ahead, we will be working intensively with our allies and other cooperating partners on measures to strengthen the international nonproliferation regime and to resolve a number of issues held in abeyance during INFCE.

As world events have reminded us repeatedly during the last three years, nuclear proliferation remains an urgent problem of international security. It seems unlikely that the problem will diminish in the years immediately ahead. Proliferation is, as we all know, a complex and difficult issue. It requires a whole range of political, strategic, and technical approaches, no one of which, by itself, can be expected to offer a "fix."

The present Administration, while not neglectful of the other dimensions of the problem, has sought to draw attention to the potential proliferation risks posed by widespread movement toward reprocessing, commerce in plutonium fuels, and the spread of sensitive nuclear facilities. It was the perception of these risks, among other reasons, that prompted the United States to revise its domestic nuclear strategy and to urge other nations to exercise restraint in premature moves toward plutonium fuel cycles and the export of sensitive technologies. The United States also took the initiative in proposing the INFCE study to explore the economic, technical, and institutional aspects of fuel cycle strategies in the light of nonproliferation and nuclear energy objectives. Domestically, the DOE's NASAP studies examined similar questions with particular reference to U.S. nuclear program choices.

The results of these studies, as well as recent world events, have confirmed that nuclear fuel cycle developments can affect proliferation risks and that this fact must be given adequate weight in national and international nuclear energy planning. The INFCE results are indeterminate on many important issues. They were arrived at through a process of technical consensus and are not fully supportive of any one strategy or national position. Still, it is a sign of growing international realism, to which INFCE made a key contribution, that although the civilian nuclear fuel cycle is not the only path to nuclear weapons development, its relevance to proliferation is more widely recognized. Therefore, though there are many questions that remain to be settled in the post-INFCE world, a return to "business as usual" without due consideration to nonproliferation issues is simply not an option.

As I mentioned, we are still assessing the INFCE and NASAP results in terms of their specific implications for DOE program choices. However, before proceeding to a discussion of the program you have asked about, I would like to mention briefly some of the key results which we believe are encouraging the U.S. hopes for restraint and caution in moves toward widespread use of plutonium and other sensitive nuclear activities.

INFCE's generally positive findings with regard to the fast breeder reactor have received considerable attention. The report of the breeder working group expresses enthusiasm about the breeder's long-term potential to realize significant uranium savings over current once-through systems. This has never been in dispute. However, it is inaccurate to say that INFCE endorsed rapid deployment of the breeder or "legitimized" reprocessing.

INFCE was conservative in its overall assessment of the breeder. It acknowledged that breeders require a major economic investment, that they may make sense only in countries with heavy deployment of nuclear power, and that their timing and need are highly dependent on the specifics of uranium supply and nuclear power demand in individual countries. Similarly, although INFCE recognized the interest of some countries in reprocessing, it also reached a number of significant statements about the benefits, costs, and risks of this sensitive technology. Although some countries see reprocessing as a positive contribution to fuel supply, it was concluded that plutonium recycle in existing reactors is not likely to have large economic advantages.

INFCE also found that safe management of nuclear wastes does not require the reprocessing of spent fuel. The long-term storage and terminal disposal of spent fuel as such, without the separation of plutonium, is technically feasible and would only require the adoption of existing technology.

Finally, it should be noted that, inter alia, INFCE explicitly recognized and endorsed the conversion of HEU-fueled research reactors to lower enrichments, the potential for considerable improvements in LWR fuel utilization efficiencies, the need for improved international safeguards, the evolution to new forms of institutional cooperation, and a need for increased uranium exploration. Mr.

Chairman, these issues are all issues which you raised in your letter to Secretary Duncan. These are areas where the Department of Energy has healthy, ongoing programs, and it is with pleasure that I have this opportunity to discuss these programs with you.

First, In addressing your general request for a review of those parts of the DOE budget which have relevance to U.S. nonproliferation efforts, I would point out that proliferation responsibilities in DOE are spread across a number of offices. Programs concerned with R. & D. on proliferation-resistant fuel cycle technologies are primarily conducted by the Office of Nuclear Energy. The NURE program is in the Office of Resource Applications, which is also responsible for uranium enrichment services. Technical safeguards issues, physical security, export controls, and nonproliferation intelligence are centered in our Office of Defense Programs. The Office of Nuclear Affairs, in International Affairs, cooperates closely with State and ACDA on nonproliferation policy matters and has important delegated responsibilities in implementing the Nuclear Nonproliferation Act (NNPA). These responsibilities include tasks such as negotiating nuclear agreements for cooperation and processing "subsequent arrangements." The Office of Nuclear Nonproliferation, under the Director of Energy Research, plays an important policy and technical coordinating role across a wide range of nonproliferation programs and issues, and administers several programs dealing with nonproliferation.

Due to the fundamental importance of the DOE programmatic and policy support for the Administration's nonproliferation efforts, these offices are not only necessary but crucial in order for us to properly fulfill our significant obligations in this most important area. I found their support to be invaluable in the role I have played in coordinating DOE's nonproliferation efforts.

REDUCED ENRICHMENT RESEARCH AND TEST REACTORS PROGRAM

As I previously noted, INFCE studies recognized and endorsed the conversion of HEU fueled research reactors to lower enrichments. The DOE fiscal year 1981 budget request for the Advanced Reactor System program is $3 million in budget authority, directed to support Reduced Enrichment Research and Test Reactor (RERTR) program activities. The objective of the RERTR program is to develop and demonstrate technology for reducing uranium enrichment in research and test reactor fuels, and to facilitate international use of such reduced enrichment fuels. The attainment of these objectives would, therefore, reduce the associated risks of weapons usable materials for research reactors by affecting a reduction in related HEU inventories at home and abroad.

The specific near-term objective of the program is to achieve the lowest feasible fuel enrichment for research reactor fuel using current technology. Fuel fabrication technology presently available in the U.S. and in Europe would permit enrichment reduction from 90 to 93 percent to 30 to 45 percent in many reactors, and to 20 percent or below in others, without significant reduction in desirable reactor performance criteria.

The long-term objective of the program is to develop high-uranium-density fuel technology which would permit enrichment reductions to below 20 percent in almost all research and test reactors. An additional and equally important long term objective is to provide the technical support necessary to make the high uranium-density fuels commercially available.

Presently, the numerous offers from reactor operators and fuel fabricators to participate in the RERTR program activities and the creation of similar RD&D programs in Japan, France, and West Germany, indicate much interest and foreign acceptance of the reduced enrichment fuel concept. Commercial fuel fabricators have already demonstrated that significant increases in the uranium-density of MTR-type fuels can be achieved by minimal modifications to current fabrication procedures. Thus, most low power-density research reactors could successfully convert to low-enrichment uranium (LEU) fuel, once such operations have been relicensed.

In fiscal year 1981, DOE is continuing: (1) the development of advanced fuels with increased uranium content, thereby facilitating further reduction of uranium enrichment requirements; (2) implementation of irradiation demonstration of reduced-enrichment fuels in selected research and test reactors; and (3) assistance to reactor operators involved in conversion of their reactors to reduce enrichment fuels.

Through DOE supported research, development and demonstration, commercial fuel fabricators should be in position by 1983 to establish commercial availability of reduced enrichment fuels which can replace the HEU fuels now being used in almost all of the low-power research and test reactors which depend on the United States for fuels. Specific activities to be carried out in fiscal year 1981 include:

1. Characterization of related neutron flux spectra and operation at design power levels suitable for full performance verification of a fuel core of plate-type LEU fuel in the Ford Nuclear Reactor (at the University of Michigan).

2. Installation of a full core of TRIGA-type LEU-fuel in the Rhode Island Nuclear Science reactor and operation at power levels suitable for fuel performance verification.

3. Continuation of advanced fuels technology development including irradiations of aluminide, oxide, and silicide reduced enrichment fuel materials and initiate post-irradiation examination and evaluation studies.

4. Continuation of technical assistance in response to request from reactor operators who are evaluating the convertibility of their reactors to reduced enrichment fuels.

5. Continuation of analysis of HEU procurement requests and of potential convertibility of reactors, as generic or specific examples.

Pledges made at the 1978 Special Session on Disarmament (SSOD) in support of the conversion of foreign research and test reactors to LEU are expected to be funded out of the Department of State's budget. As specific requests for assistance from foreign operators are identified by State, DOE will provide technical support for these conversions.

URANIUM UTILIZATION IN LIGHT WATER REACTORS

The light water reactor (LWR) will produce the bulk of the world power from nuclear energy well into the next century. Since its commercial introduction almost 20 years ago, the LWR has been making steady progress in improving safety, reliability, and fuel efficiency. Considering its prominence in nuclear power generation and its proliferation resistance when operated on the once-through cycle, DOE has instituted a long-range program to assist industry in upgrading its capability.

Mr. Chairman, you have asked that the Department specifically address the funding levels and objectives for the program to improve uranium utilization in LWR's. This important program has the following milestones for improvement of the LWR through the remainder of this century.

The Department will concentrate its efforts through the 1980's on backfittable improvements that can have a significant near-term impact on all existing and currently planned LWR's. The present high priority program concentrates on development of high burnup fuel. Successful completion of this program should provide the technology with the potential for increasing LWR fuel efficiency by about 15 percent before the end of the decade. Other backfittable improvements could be expected to add further to the increased fuel efficiency.

Advanced designs that could have commercial potential are also being considered by DOE. These concepts concentrate on more fundamental improvements in the once-through LWR that could be incorporated in a new reactor design. These concepts could lead to total uranium savings in new light water reactors of about 30 percent by the end of the century.

The Department's program for developing and demonstrating uranium utilization improvements has four major parts:

1. Development and demonstration of higher burnup fuel—this is the most important near-term method for substantially improving uranium utilization;

2. Development of other backfittable improvements (those which can be readily utilized in existing plants);

3. Supporting research and development to provide the technical basis for high burnup fuel design; and,

4. Examination of nonbackfittable reactor design features (those which involve major plant changes and are, therefore, practical only in new plants).

The high burnup fuel program has a number of important side benefits. First, the continued evolution of LWR fuel efficiency will reduce the demands on uranium resources and production capabilities to a significant degree. Second, the increased burnup from current levels to the DOE target of 50,000 MWd/MTU would reduce the amount of spent fuel discharged by about 40 percent. The improvement program, particularly the advanced LWR program, has the potential to reduce LWR separative work requirements in the long term.

Substantial interest in improving uranium utilization in light water reactors is being shown in a number of other countries as well. This interest is evidenced by a willingness to participate in multilateral cooperative projects. One such project is the high burnup effects program being organized by Battelle-Pacific Northwest Laboratories. This project will evaluate the effect of fission gas released in high burnup fuel. Both the Japanese nuclear industry and the European nuclear industry are participants. The European countries include Britain, Germany, Sweden, Denmark, Finland, Italy, Belgium, The Netherlands, France, and Switzerland. Several other international research and development projects are currently in the formulative stages.

The funding requested for the LWR uranium utilization program for fiscal year 1981 is $21 million. This is an increase of $3 million over the fiscal year 1980 budget.

PROLIFERATION RESISTANCE OF PRESENT AND FUTURE NUCLEAR SYSTEMS

Mr. Chairman, you requested that we discuss proliferation resistance of both present and future nuclear systems. For current systems, the Department's Nonproliferation Alternative Systems Assessment Program (NASAP—addressed later in this testimony) has concluded that the LWR operated on the once-through fuel cycle is a relatively proliferation resistant commercial power technology in that directly weapons usable material does not occur in its fuel cycle. The DOE has programs under way to further improve the LWR and enhance its long-term viability.

Considering future systems, the most significant improvements which we believe could be implemented at an early date are improved international safeguards and the continued development of institutional arrangements among breeder developing nations. Improved safeguards for LMFBR systems are currently being developed in the United States to identify means for increasing the security of all aspects of the fuel cycle including handling of fissile material, fabrication of fresh fuel, transportation of fuel, handling and control of fuel in the reactor plant, handling and storage of spent fuel, and reprocessing. We are coordinating our efforts in this area with other breeder developing nations and our ultimate objective is to develop a set of international safeguards which could be uniformly implemented through the IAEA.

Technically, the Department is also exploring breeder improvements that could be more proliferation resistant through a decrease in the frequency of fuel reprocessing and reduction in the amount of plutonium that may exist outside the reactor core at any given time. These improvements are associated with the extended burnup that may become possible with high breeding ratio FBR cores.

With respect to the LMFBR, proliferation resistance can be enhanced by reducing the requirements or frequency for reprocessing. This can be accomplished by extending the burnup capabilities and hence lifetime of oxide or carbide fuel systems. Such a program is under way and indications are that lifetimes, significantly beyond the current 80,000–100,000 MWd/t range, can be achieved.

Another approach that has been considered involves a concept referred to as the Fast Mixed Spectrum Reactor (FMSR). This concept also addresses the absence of or use of infrequent reprocessing by extending the burnup capability. It also proposes the use of metallic-type fuels utilizing large quantities of low enriched uranium, a varying neutron spectrum, either sodium or helium as a coolant, and fuel residence times on the order of 15 to 20 years. The feasibility of this approach has, however, not been proven, and the performance requirements are beyond the known or projected capability of materials being developed or examined to date.

Specific efforts on Advanced Nonproliferation Reprocessing Technology have been considered in fiscal year 1980. Work in this subprogram will be directed toward reprocessing plant design concepts which would virtually eliminate direct access, in the fuel reprocessing operations, to materials usable in nuclear weapons. These concepts involve zero access to process cells through the use of complete remote operation and maintenance of reprocessing operations as well as remote, in-cell analytical sampling procedures. Process-control systems could be expanded to incorporate the surveillance of diversion-attempt information and passive use-denial actions by the use of a command, control, and communication network.

a wide range between the low demand estimates and the high demand estimates and the same thing is true worldwide.

What you ultimately decide is going to be the figures that you hang your hat on is a matter of judgment. I think people come to different conclusions on that. INFCE is a case where you have a lot of different ideas about what the world is going to look like in the long run. I think the way that subgroup proceeded reflected a consensus about what that world would look like.

I think it is different than our own projections. This is a consensus process of technical representatives. I think it is not unlikely that you are going to find significant differences in those outlooks.

Mr. NOSENZO. I might add to that, that these estimates were 1968 estimates and as you know the key contributors to those estimates were the IEA countries, the OECD–IEA countries. The estimates that were used in INFCE were in fact the IEA estimates.

I do not know if you know how these estimates are formulated. IEA countries fill out questionnaires giving their nuclear projections and submit them to the IEA secretariat. They then go through a very careful scrubbing. The IEA secretariat reviews each one and asks various questions of each country to get the best possible estimate, recognizing that country's projections really represent their goals rather than what they will realize.

If you were to make these estimates right now based on the current IEA projections, you would get a much lower value. To characterize it as a U.S. projection compared to an INFCE projection isn't really accurate. I think it is more characteristic of what the global projection would be today versus what it was in 1978.

RERTR PROGRAM

Mr. WOLPE. Let me switch to a discussion on the reduced enrichment research and test reactor program. I understand Iraq is importing a large scale research reactor as well as the highly enriched uranium which is necessary to fuel it. This is obviously a worrisome development for the security of that region.

Could you tell us whether the fuel being developed under the RERTR program would eventually help to reduce concerns such as those posed by a case like Iraq where highly enriched fuel must be sent because no alternative fuel form is available?

Mr. BATEMAN. That certainly is the intent. The question is over what period you can accomplish this. It is clear that the types of LEU fuel which are nearest to deployment are for low power reactors and the ability to develop that fuel and to make it commercially available probably will take place over the next 3 years, say sometime in 1983 or at least in that range.

For low-power reactors I think the picture is fairly clear and I think optimistic. For the higher power reactors and the more advanced fuel technologies you are really talking about a longer range picture perhaps a 10-year period at the outside.

I think it depends on what kind of reactors you are talking about and what kind of near term or advanced fuel development technology is involved.

Mr. WOLPE. Nevertheless the Iraqi example would fit the goal of the program?

Mr. BATEMAN. Yes.

BUDGET CUT

Mr. WOLPE. I understand the administration's $5 million fiscal year 1981 request for this program has been reduced to $3 million. Could any of you gentlemen indicate how it is possible to justify a cut of that sort in a small but vitally important security-oriented research effort when there are so many hundreds of millions of dollars left untouched in broader fuel cycle research?

Mr. BATEMAN. The intent throughout in terms of the budget cutting exercise was not to make any long-range programmatic cuts in these budgets. We are talking about cuts which affect only 1981, in an attempt to balance the budget in that year.

It is not correct to read the $5 million to $3 million cut as a change in our long-range programmatic goals. We believe the $3 million is consistent with meeting our near-term objectives with an acceptable schedule slippage in the years beyond 1981; we think that it will permit meeting our long-range programmatic objectives, in terms of developing these advanced fuel types.

I just want to stress that the budget cutback is not intended in this program or the others to reflect changes in long-range programmatic goals of the administration. We are committed to this program as we have been.

Mr. WOLPE. In the short range could you indicate specifically what will be the impact of the proposed cut on DOE efforts to develop safer research reactor fuels?

Mr. BATEMAN. In the long run I think the impact will be an extension of the program by about 1 year. In the short run it will retard fuel development activities and will delay work on the development of advanced fuel types for 1 year or so.

Mr. WOLPE. Thank you.

Mr. BINGHAM. Mr. Gilman.

Mr. GILMAN. Thank you, Mr. Chairman.

I would like to pursue this issue a little further on reduced enrichment research. If additional funding were made available, how could it be utilized to make this program more effective?

Mr. BATEMAN. Basically on work having to do with advanced fuel technology development for these larger research reactors and on expediting the stretched out R. & D. activities currently planned.

Mr. GILMAN. Do you have some specific programs that you would utilize it within that broad range?

Mr. BATEMAN. Yes, sir.

Mr. GILMAN. When it was reduced from $5 million to $3 million, did you take part in that reduction recommendation?

Mr. BATEMAN. No, sir.

Mr. GILMAN. How would the additional $2 million have been utilized?

Mr. BATEMAN. On advanced fuel technology development activities.

Mr. GILMAN. Are there some projects right now that are underway with regard to advanced technologies?

Mr. BATEMAN. Yes and those would be stretched out for at least a year along with stretchout of near-term fuels development activities under this revised budget.

EXHIBIT E

PROGRESS IN U.S. AND INTERNATIONAL NONPROLIFERATION EFFORTS

HEARING

BEFORE THE

SUBCOMMITTEES ON INTERNATIONAL SECURITY AND SCIENTIFIC AFFAIRS

AND ON

INTERNATIONAL ECONOMIC POLICY AND TRADE

OF THE

COMMITTEE ON FOREIGN AFFAIRS

NINETY-SIXTH CONGRESS, House

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MONDAY, MARCH 12, 1979

HOUSE OF REPRESENTATIVES,
COMMITTEE ON FOREIGN AFFAIRS,
SUBCOMMITTEES ON INTERNATIONAL SECURITY AND
SCIENTIFIC AFFAIRS AND ON
INTERNATIONAL ECONOMIC POLICY AND TRADE,
Washington, D.C.

The subcommittees met at 3 p.m. in room 2200, Rayburn House Office Building, Hon. Clement J. Zablocki (chairman of the committee) presiding.

Chairman ZABLOCKI. The subcommittees will please come to order. We meet today to consider technologies and programs that can help make the nuclear fuel cycle and the U.S. nuclear exports more safeguardable and secure. As Dr. Van Doren kindly notes in his prepared statement, this committee has been in the forefront of those trying to assure timely warning of any diversion of nuclear materials well in advance of the time it would take to transform such material into nuclear explosive devices.

I wish to commend the representatives of the Departments of State and Energy and ACDA for their work in pursuit of this goal and for the efforts they have already taken to restructure nuclear research and development along more proliferation-resistant lines. Despite much initial skepticism, these labors have already begun to bear fruit as demonstrated by our success today in reducing the enrichment of research reactor fuels. As for future alternatives, we are of course uncertain. But the technologies you are promoting today are clearly important in that they help to buy time, time in which safeguards and research can continue; new designs explored; economic and resource uncertainties further narrowed; and most importantly, time during which the international community can prepare to deal with security problems for which no immediate answers exist.

At least now, for the first time, the question of how to reduce proliferation dangers is being considered, before, not after, nuclear technologies are undertaken. This is an important step.

This is a joint committee meeting of the Subcommittee on International Security and Scientific Affairs and the Subcommittee on International Economic Policy and Trade. I call upon the chairman of the latter subcommittee to welcome you, gentlemen.

Mr. Bingham.

Mr. BINGHAM. Thank you, Mr. Chairman. I would simply note that just about a year has passed since the President, with the strong sup-

port of Congress, set in motion the Nuclear Non-Proliferation Act of 1978. During that time the United States has raised international awareness of the inherent security risks posed by the commercial nuclear fuel cycle. Concrete steps have been taken to mitigate these risks, including decisions by West Germany and France to join the United States in suspending further exports of nuclear reprocessing equipment. Over 50 nations are participating in the international nuclear fuel cycle evaluation, INFCE, and we are particularly anxious to hear about that today. Together we are reviewing all aspects of the nuclear fuel cycle, with specific focus on technological and institutional means of reducing proliferation.

This is the first of a number of planned hearings this year on progress in international and U.S. nonproliferation efforts, and we look forward to hearing the three agenies represented here today.

Thank you, Mr. Chairman.

Chairman ZABLOCKI. Ambassador Pickering, Mr. Deutch, and Mr. Van Doren, we welcome you back to the committee. We are pleased, once again, to have the benefit of your judgment on these important matters. We have prepared statements that you have filed with the subcommittees. You may either read the entire statement or summarize it as you wish. Your entire statement will be made part of the record.

Ambassador Pickering, would you begin?

Mr. PICKERING. Mr. Chairman, since the basic focus is technical measures and some institutional measures, I wonder if we could begin with Mr. Deutch.

Chairman ZABLOCKI. I understand further that you have a time problem.

STATEMENT OF HON. JOHN M. DEUTCH, DIRECTOR OF ENERGY RESEARCH, DEPARTMENT OF ENERGY

Mr. DEUTCH. Mr. Chairman, I am pleased to be here. Thank you very much.

With your permission, sir, I will present my prepared statement for the record, including answers to specific questions raised in the letter sent to the Department inviting us to appear, and just summarize my remarks, if that is agreeable with you, sir.

Chairman ZABLOCKI. Without objection, your prepared statement of the answers to the questions that were supplied will be made part of the record.

Mr. DEUTCH. Thank you. I am very pleased to appear here this afternoon to discuss the status of the international nuclear fuel cycle evaluation and, more particularly, the Department of Energy technical programs, broadly in support of the U.S. nonproliferation policy.

I should like to mention that I am personally committed to the success of that policy, because I regard it as being essential for the national security of the United States, and world security in general. I believe that the Department of Energy in all its different places, and forms and organizations is working deliberately to strengthen our nonproliferation policy.

It has been my observation that experts come to the problem with different points of view. Lawyers and diplomats look for technical solutions, while technologists look for institutional solutions. I think we all recognize that there is not going to be a single moment where there will be a magic set of arrangements, whether they are technical or institutional, that will lead to an ultimate resolution of this problem. It is something we will have to work at with great ingenuity and perserverance, using both technical and institutional measures in order to make progress.

DOE ORGANIZATION

I would like to describe briefly for you the organization of the Department of Energy in support of the international nuclear fuel cycle evaluation. As was mentioned by Chairman Bingham, there are 53 nations and 4 international organizations involved in the international fuel cycle evaluation. The leadership for our Nation's participation in this important activity comes under the Department of State through Ambassador Gerard Smith and Assistant Secretary Pickering. There is a management committee which is composed of representatives of the different concerned agencies, including two members of the Department of Energy, who work together to formulate a joint U.S. position.

We are about at the midpoint of the international nuclear fuel cycle evaluation. By the end of this fiscal year the Technical Coordinating Committee, which is charged with integrating results of the international nuclear fuel cycle evaluation, will be taking reports of eight different working groups and bringing them together for consideration by the plenary body.

The Department of Energy's support to the international nuclear fuel cycle evaluation comes in a variety of forms. We have an international nuclear fuel cycle coordinating office at the Department of Energy. That office is responsible to the Director of Energy Research—that is me—and it is headed by Mr. Eric Beckjord. He is the principal point of contact between the Department of Energy and the Department of State on these matters.

We have over 40 professionals in the Department of Energy, technical experts of one type or another, that are participating in the preparation of papers or in the deliberations of the eight different working groups that are ongoing in the international nuclear fuel cycle evaluation study. Our support, our financial support for the U.S. participation in this technical work and in the work of these meetings is not earmarked specifically in the Department of Energy budget. Rather, it comes from our nonproliferation alternative systems assessment program, which supports a variety of U.S. studies designed not only for use in the international nuclear fuel cycle evaluation but in addition, to determine our own reactor development strategy in this country. That program in fiscal year 1979 will be spending $6.9 million. The total estimated cost of the program is expected to be in the range of $24 million through fiscal year 1980.

It is important because it provides a common analytic basis on which to evaluate fuel cycles and reactor systems, and the relative proliferation risks that they may entail.

TECHNICAL RESULTS OF INFCE

Next I would like to speak about the preliminary technical results that are emerging from the international nuclear fuel cycle evaluation program. We all agree that there will be no magical technical fix, and we also all note that it is premature to speak about all the results that may emerge from our efforts since we are only at the halfway point in the study. However, I would like to highlight a few of the major technical thrusts that appear to be getting widespread support.

The first is the once-through light water reactor system. We can improve its fuel utilization. It is the correct baseline to use for assessment of proliferation resistance. This once-through light water reactor system is at the present time, in our judgment, the most proliferation-resistant fuel cycle we know. We may discover that the risks in alternative fuel cycles can be minimized, can be changed, can be reduced. However, we must recognize that the standard of measurement is the once-through light water reactor system.

Second, we have recognized that there have been in international commerce a large number of research reactors which have been fueled by highly enriched uranium. Indeed, there are approximately 5 metric tons of this fuel in the cores of research reactors in the free world, including the United States. The general conclusion emerging from INFCE after initial questioning is the desirability of developing alternative fuels which will permit the same physics to be done in these research reactors, in as safe as possible a way, while reducing enrichment to, hopefully, 20 percent but at least 45 percent. At the end of a program of conversion we believe there would be no more than five or so reactors in the world which would be required to still use highly enriched uranium.

Finally, speaking of the preliminary results coming out of INFCE, we do recognize, and it was broadly recognized in other participating countries, the importance of at least examining alternate fuel cycles and alternate breeder concepts. While I would not wish to give you the impression that all nations are immediately agreeing that there are better fuel cycles than the plutonium purex reprocessing system, or that there are breeders which might be more proliferation resistant than liquid metal fast breeders, we do believe there was general agreement on the need to examine alternate fuel cycles and alternate breeder concepts. Furthermore, we need to examine certain measures that would mitigate the proliferation risk of the present or alternate fuel cycles that are under examination.

Your letter, Mr. Chairman, requested information on precise, quantitative proliferation criteria. I might note that within INFCE, five countries—the United States, the United Kingdom, West Germany, France, and India—have provided papers on proliferation-resistance criteria. I think that you must all recognize that there will not be simple quantitative criteria available, agreed to by all, by which to judge proliferation resistance.

THREE CATEGORIES OF CRITERIA

However, there has been general agreement that there are at least three categories of criteria that require examination. The first is resource requirements for proliferation. These resource requirements are of three types: Dollars, that is, the amount of money that it would take to misuse the fuel cycle; people, the number of trained technical people and total people that are required to misuse the fuel cycle; and finally, technological complexities, how complicated is it for a certain nation, given its scientific infrastructure, to misuse an installation or a fuel cycle.

The second proliferation criterion that has gained wide acceptance concerns the time needed to acquire strategic nuclear materials. I must say that I have personally been pleased to see that technical people from a wide range of different countries not only agree that the question of time to diversion of certain kinds of technology and certain kind of plant is of importance, but beyond that, to see that their calculations of time closely agree in given circumstances.

The third proliferation criteria which is of great importance concerns detectability, both of covert operations and of diversion from safeguarded facilities.

I would like to next turn to the central programs that the Department of Energy has underway which I believe strongly support our nonproliferation policy. It should come as no surprise that the core of these programs has to do with improving and extending the lifetime of our light water reactor system in the United States. This, we believe, is extremely important to our nonproliferation objectives, as well as to the questions of energy supply in this Nation.

There are a variety of programs that I would like to touch on that bear on this effort. First is our new program this year for light water reactor improvements. In this program we have four different efforts, all of which are aimed at making light water reactors a stronger and more extended option for the United States. We have efforts going on in reactor safety; in the reliability of reactors, that is, keeping their capacity factor high; and in reducing the radiation which people who work around reactors are exposed to.

IMPROVING URANIUM EFFICIENCY

Finally, the fourth major effort in our light water reactor improvement program is an aggressive effort to improve the efficiency of reactors in their consumption of uranium. We believe that improved fuel management and higher burnup can lead to a 15-percent increase in the economy of reactor uranium usage of the existing generation of reactors.

In addition, we believe that there is a 10-percent to 15-percent additional improvement that may be possible, and I would like to emphasize here that good intentions are not enough. One must run a technical program to assure that this additional amount of efficiency is possible, sometimes retrofittable in an existing generation of reactors and sometimes not. But there will be an additional 10 percent to 15 percent for uranium efficiency for our light water reactors.

Of course, this is of paramount importance because if we can reduce the lifetime uranium requirements of a reactor to produce a certain amount of power, it permits the existing ore base to cover a longer lifetime of a reactor inventory, or more reactors operating for a given period of time without the need to enter a plutonium economy.

It is not only important for the United States, but it is important for other nations of the world who wish to take advantage of nuclear power, but whom we do not wish to encourage to make a premature commitment to reprocessing for thermal recycle.

This program of improving uranium efficiency, which is one part of our light water reactor program, is slated for $18 million in fiscal 1980 in contrast to $14 million in fiscal 1979. Perhaps the greatest advantage that all of these programs taken together to improve light water reactor performance in the United States is that it will provide not only for reliable power which does not bear great proliferation risks, but also for less expensive power to the American consumer.

ISOTOPE SEPARATION PROGRAM

There are many other programs that bear on our nonproliferation policy. We next turn to our advanced isotope separation program for which we are requesting $55 million in fiscal 1979. The advanced isotope separation program, particularly laser isotope separation, has two advantages associated with it. First of all, it is an economic way of stripping tails; that is, removing the last amounts of uranium-235 from the depleted tails that are presently being stockpiled after leaving enrichment plants for use in reactor fuels. This is something that will also extend the lifetime of the light water reactor system in this country.

Equally important, our advanced isotope separation technology efforts will continue to place the United States in a position of being a reliable supplier of enrichment services, in particular, a reliable supplier of enrichment services at a low price, so that our allies will be able to look to us for the provision of reliable enrichment services at a price which is competitive.

NURE PROGRAM

The third program that bears very heavily on our nonproliferation policy efforts is the NURE program, the national uranium resource evaluation program. We are requesting $80 million for fiscal 1980. This program is directed toward assessing the amounts and the costs of domestic uranium ore supplies. As our knowledge of the supply curve for uranium improves and our knowledge of the resource base improves, there will be greater confidence in staying with the light water reactor system for a longer period of time.

I would next like to mention our efforts with alternative fuel cycles and with alternative reactor systems. I would, first of all, mention that we are continuing this year at a reduced budget level our research and development into the high temperature gas reactor HTGR, particularly in the use of low- and medium-enriched fuel for the charging of such reactors. We are not proceeding at this stage in reprocessing of HTGR fuel. We are proceeding with a longer look at direct cycle applications of the HTGR. This is a particularly important program with our allies, particularly with the Germans, who have a great interest in developing alternatives to the light water reactor system that has some advantages, particularly in the reduced use of water

for operating reactors. In some parts of the world the water requirements for reactors are of considerable importance.

ALTERNATIVE FUEL CYCLES

We are also continuing to examine alternative fuel cycles and reactor types. We have modest programs underway for matters such as coprocessing, using partially decontaminated fuel after reprocessing, and a particularly important effort underway to examine what engineering features can be added to conventional purex reprocessing to make that rather vulnerable technology more proliferation resistant.

I do not wish to dwell at this time on our breeder program, unless members of the committee have specific questions. Let me say that our liquid metal fast breeder program is not only looking at plutonium/uranium systems, but also at thorium/uranium-233 systems. We are continually looking at all breeder technologies such as gas-cooled reactors, molten salt reactors, accelerator breeders, and even hybrids. In all of these cases, we have not determined at the present time that there is an alternate breeder concept to the LMFBR that offers qualitative proliferation advantages.

RESEARCH REACTOR PROGRAM

I would like to conclude, Mr. Chairman, with a brief discussion of the research reactor program we have requested in this year's budget. I believe we have requested $8.3 million in fiscal year 1980 for our efforts in reducing the enrichment required for research reactors. We are in the process of changing that request to $5 million.

I would like to make a remark about our efforts on this very important research reactor conversion program. There are two steps which are required. The first is a near-term step that will take between now and about 1982, which will require that we develop alternate fuels of lower enrichment and demonstrate that high-density fuels can be used in research reactors with adequate safety, and yet provide the physics for which those research reactors were built.

In the development of these fuels, they are tested, and then their demonstration in the reactor is a part of the program under which we are currently embarked and for which my Office of Energy Technology is responsible. We believe that it will take up to the end of 1982 before these fuels have been demonstrated and there are radiation tests which supply completely compelling information for reactor operators. For safety and for physics considerations it will take until the end of 1982. The total cost of that program will be about $27 million.

Subsequent to that we will have to undertake a deployment program for this medium or low-enriched fuel in our reactors. This deployment program will consist of two parts. In the first place it will be necessary to convert our U.S. domestic research reactors. There are two reasons for this. First, to show that it is possible to convert research reactors and to set an example for other nations of the world. Second, to establish the required experience which will be needed to guide our further efforts.

DOMESTIC RESEARCH REACTORS

There are two types of domestic research reactors in this country: There are domestic research reactors that are basically run by the Department of Energy or by universities under Department of Energy contract. The Office of Energy Research, of which I am the Director, is in the business of buying fuels for such reactors. The pace and the cost at which we convert these reactors, in particular those reactors which contain more than 1or 2 kilograms of highly enriched uranium, is one that will have to be decided on in the future.

We will be, during fiscal 1979, laying out a specific program for the deployment of the technology that will be finally available at the end of fiscal 1982.

The second class of domestic reactors concerns those which are in the hands of private industry. The methods that we use to encourage their conversion is something we will also have to be paying attention to during fiscal 1979.

CONVERSION OF FOREIGN RESEARCH REACTORS

Finally, we have the question of our encouraging conversion of foreign research reactors, We will be working closely with the Department of State and the Arms Control and Disarmament Agency to decide what is the best strategy for employing these reduced enrichment fuels. I want to say to you that I am extremely optimistic and have found a great deal of interest in other countries. Both Germany and France, for example, are not only fabricating such fuels themselves, but also assisting in their deployment in the world today.

We in the Department of Energy strongly favor the objectives of this program. It is our responsibility to undertake the technology development and demonstration in a timely and effective way, and we are so proceeding.

Mr. Chairman, that is a brief overview of the major programs underway among our nuclear programs to support nonproliferation objectives, and the International Nuclear Fuel Cycle efforts in particular.

With that I would like to close and apologize for perhaps having gone on too long. Thank you very much, Mr. Chairman.

Chairman ZABLOCKI. No; it is a very technical subject, and you can't do it in just a few paragraphs or a few pages. I think you did very well, not only in your prepared transcript. I must say, you did remarkably well.

Mr. DEUTCH. Thank you, sir.

[Mr. Deutch's prepared statement follows:]

PREPARED STATEMENT OF HON. JOHN M. DEUTCH, DIRECTOR OF ENERGY RESEARCH, DEPARTMENT OF ENERGY

Messrs. Chairmen and members of the subcommittee: I am very pleased to appear before you this afternoon to discuss the status of the International Fuel Cycle Evaluation (INFCE) study and the Department of Energy's technical programs in support of U.S. nonproliferation policy. As you know, an important part of that policy is the effort to develop nuclear fuel cycle approaches that are as proliferation-resistant as possible. This goal is and must remain a central consideration as we plan our own nuclear energy strategy and as we work with other nations to develop a consensus for a safer global regime for nuclear energy development. At the same time, however, it is important to view these technological efforts in a larger context of nonproliferation policy generally and to recognize that there are no "technological fixes" that in themselves can guarantee us a risk-free nuclear future.

My observation is that the experts come to the problem with different points of view: lawyers and diplomats look for technical solutions, while technologists look for institutional solutions. In fact, U.S. nonproliferation policy is, as it must be, a blend of different approaches, including day to day diplomatic as well as longer term, institutional, economic, and technical elements. Moreover, we have recognized that many of the basic incentives towards developing nuclear weapons necessarily can be dealt with only at a political level whether a nation takes the final step is heavily dependent on its sense of political and military security, and its technical and economic capabilities.

Having said this, however, we have recognized that there is a strong technical component to the problem. The risks of proliferation could be seriously aggravated by the uncontrolled spread of sensitive materials and facilities or by a situation in which our institutional safeguards and controls are not judged to be fully adequate to deal with the quantities of weapons-usable materials that may be readily accessible in the fuel cycle. Factors such as these have prompted the United States, in the past, to push vigorously for the widespread acceptance of the Nonproliferation Treaty and the stengthening of IAEA safeguards. In addition, however, these considerations prompted the Carter Administration to take a substantially harder look at the proliferation issue from the technical standpoint.

We believe that the technological innovations can broaden our choices, not only technically but also politically and institutionally, in a way that strengthens the proliferation-resistance of nuclear energy development. This is not only true for the long-term as we develop follow-on systems to our current generation of nuclear power. The improvements we can make in the near- and mid-term in existing reactor systems can extend the resource base and hence the lifetime of the relatively proliferation-resistant Light Water Reactor (LWR) once-through cycle. This in turn can buy us time before any irrevocable commitments to more sensitive fuel cycles are necessary, improving the chances for developing needed diplomatic and institutional frameworks in the interim. In this way, technological improvements lend crucial support to the evolutionary approach to nuclear development that we have advocated.

In recognition that the United States could not embark on major new nonproliferation initiatives alone, we called for the inauguration of a major new International Nuclear Fuel Cycle Evaluation (INFCE). The purpose was to enable the nations of the world to pause and systematically consider the principal options that might be most supportive of nonproliferation objectives. The United States entered the analysis without fixed preconceptions but with a determination to explore approaches that might minimize the presence of weapons-usable materials while assuring the orderly growth of nuclear power.

INFCE

The INFCE Organizing Conference held in October 1977, in which 40 countries participated, set the purpose of INFCE in the following terms:

"The participants in the Organizing Conference of the International Nuclear Fuel Cycle Evaluation are conscious of the urgent need to meet the world's energy requirements and that nuclear energy for peaceful purposes should be made widely available to that end. They are also convinced that effective measures can and should be taken at the national level and through international agreement to minimize the danger of the proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes."

INFCE's work in fiscal year 1978 was focused on scoping efforts, and on collecting, organizing and analyzing data and information needed for the evaluation. To date, over 200 international papers have been produced. Several of the eight working groups are now well advanced in drafting sections of the final working group reports.

By the end of fiscal year 1979, it is expected that each of the groups will have completed reports for consideration by the final Technical Coordinating Committee (TCC) and Plenary meetings. In addition, the TCC is preparing a summary document on the working group studies and providing an overview of the INFCE

work. The overall work of INFCE is proceeding on schedule towards completion in February 1980.

The United States contributions to and participation in INFCE are coordinated among the concerned agencies in the Executive Branch by the Office of Ambassador Gerard Smith, Special U.S. Representative for Non-Proliferation Matters. There are three principal agencies supporting our involvement: the Department of State, the Department of Energy and the Arms Control and Disarmament Agency. Several other agencies contribute expertise as appropriate including the Nuclear Regulatory Commission, the Environmental Protection Agency, and Council on Environmental Quality. Representatives of the United States are actively involved in all INFCE meetings.

The Department of Energy provides most of the technical data used in the U.S. contributions to INFCE drawing upon resources available, mainly from its Office of Energy Technology and also from International Affairs, Resource Applications, Energy Research, Environment, Defense Programs, Energy Information Agency, and Policy and Evaluation. More than 40 professionals from these organizations have been working on a part-time basis as active participants in eight U.S. support groups and several cross-cut groups. Three full time DOE professionals make up the INFCE Coordinator's Office. Two DOE people serve on an INFCE Management Committee.

In addition to those professionals providing direct support to INFCE activities, about 7 professionals support the Non-Proliferation Alternative Systems Assessment Program (NASAP) which is the source of most technical data and analysis used by the United States in INFCE activities. The overall goal of NASAP is to recommend strategy options for implementing civilian nuclear systems which, when deployed in the United States and internationally, offer increased resistance to proliferation, while maintaining the benefits of nuclear energy over the long-term. This includes the screening of all candidate nuclear power systems and the selection for detailed analysis of the most promising options, that is, those that offer high proliferation resistance, are technically and economically feasible, have high promise for commercial introduction, have high expectation for efficient resource utilization, and have acceptable public health and safety and environmental characteristics.

INFCE as such has no line item in the budget, but rather draws upon various DOE budgets, including NASAP. Fiscal year 1979 budget authority for the NASAP program, which is expected to be completed in December 1979, is $6.9 million. The total estimated cost of the program is $24.2 million.

In addition to the NASAP program, other ongoing U.S. nuclear program activities have been providing applicable data and analyses for INFCE. These include programs for development of thermal reactors, breeder reactors, advanced reactors, fuel cycle technology, enrichment, and waste management.

Since INFCE and NASAP are not yet complete, research and development program recommendations based on their conclusions would be premature. However, some preliminary findings are emerging which are helping to shape our view of future programs. Briefly, these include:

As mentioned at the outset, there is no pure technical or institutional "fix" which, applied alone, will eliminate the risk of proliferation.

The once-through fuel cycle is the most proliferation resistant of systems studied so far.

The LWR on the once-through cycle with improved fuel utilization is the preferred converter reactor.

Most research reactors designed to use highly enriched uranium (HEU) can be redesigned to use medium or low enriched uranium (MEU or LEU).

These preliminary findings have led to the program emphasis on improving light water reactor fuel utilization and the development of low and medium enriched research reactor fuel. In addition, alternate fast breeder reactor fuel cycles which produce excess denatured fuels for use in thermal reactors are part of the research and development program. Means of making reprocessing less vulnerable to proliferation, such as coprocessing, have become part of the U.S. R & D. program and are possible candidates for INFCE recommendations.

The principal technical alternatives being advanced by the United States in INFCE include fuel utilization improvements in light water reactors, lower enrichment fuels for research reactors, and alternate fast breeder reactor fuels and fuel cycles. The Department believes that these alternatives are all technically feasible and potentially attractive from the standpoint of economics, resource utilization and safety and environmental impacts. However, research and development is necessary to prove out their validity.

When the INFCE report is complete, and NASAP studies are finished, the DOE will be prepared to make specific program recommendations.

In regard to establishment of meaningful and widely accepted criteria for proliferation resistance, progress has been made in INFCE in sensitizing the international community to the urgency of this problem. Five countries (United States, United Kingdom, the Federal Republic of Germany, France and India) have submitted papers on either proliferation risk assessment or criteria to INFCE, largely as a result of U.S. initiatives. Although it is too soon to say that a consensus has emerged, the TCC has requested that the working groups perform an assessment of proliferation risk for each element of the nuclear fuel cycle. The results will be included in the working groups' reports and overview.

The U.S. initiatives which stimulated this response suggested three criteria for proliferation risk assessment of a fuel cycle activity. These are:

(1) the resources required to extract fissile material from the fuel cycle which could be used for the purpose of making weapons, i.e., manpower, technology and investment;

(2) the time needed to complete the job, from commencement of the activity until the production of sufficient material for one or more weapons, and

(3) the detectability of the activity, that is to say, the means and degree of difficulty by which the international community could become aware of the specific activity.

Much discussion has taken place on this general subject, and also on specific fuel cycle activities. One method that has been considered is to take a quantitative approach, determining specific measures for the various parts of the activity, and concluding a value for the risk. Another method is to examine activities on a case-by-case basis to render a qualitative conclusion on proliferation risk. The predominant view is that a qualitative approach is required, and that quantitative assessments are misleading and likely to become outdated as technology and skills improve and become more widely dispersed in the future.

I would now like to turn to a review of our LWR, advanced systems, and research reactor conversion programs. In doing so, I will attempt to highlight for you the relevance of each program to our nonproliferation objectives.

LWR FUEL UTILIZATION AND FUEL CYCLE ENHANCEMENTS

Light water reactors (LWRs) operating on a once-through fuel cycle appear to be the best reactor system for meeting projected near-term nuclear growth with acceptable proliferation characteristics. If LWRs are to play the significant role expected of them, it is essential that adequate uranium be available to fuel them. To help assure this, DOE has two technology efforts and one information effort underway: the Uranium Utilization Program which will improve efficiency of uranium use in once-through LWRs; the Advanced Isotope Separation Technology (AIST) Program to extract more fissionable uranium from uranium mined; and the National Uranium Research Evaluation (NURE) program to determine and possibly extend the uranium resource base.

Taken together, the Uranium Utilization and AIST programs could potentially result in a safe, reliable nuclear energy supply using from 60 to 65 percent of the amount of uranium currently consumed by present systems per megawatt of power generated. If our program results in technologies that are economically attractive, the impact of the evolution of a fuel-efficient LWR once-through fuel cycle will be substantial, not only on the preservation of a viable nuclear energy option, but also on proliferation concerns. First, it can lessen the perceived need for thermal recycle of plutonium and delay or avoid the resulting "plutonium economy." Second, it will reduce demand for uranium supplies, relieving the pressure on escalating uranium prices. Third, it will provide more time for the orderly establishment of breeder reactor programs to handle the long-range fuel supply problems and minimized attendant proliferation risks.

It should also be noted that LWRs can assume a larger share of our energy production by further increasing their reliability and enhancing their public acceptance. Therefore, in addition to improving the uranium utilization efficiency of LWRs, we are proposing a complementary objective of LWR operating performance and safety improvement. Successful completion of this program will assure that LWR plants will deliver their full potential of energy generation.

Uranium Utilization.—The Uranium Utilization effort is being carried out as part of the Department's Thermal Reactor Technology Program. The budget request for this activity for fiscal year 1980 is $18.0 million in budget authority compared to $14.4 million in fiscal year 1979.

The objective of the program is to develop and demonstrate technology which can potentially reduce LWR uranium requirements by 15 percent by 1988 and possibly up to 30 percent beginning in the year 2000.

The fiscal year 1980 funding requested for this activity will be directed primarily at achieving the near-term objective. Initial emphasis is being placed on developing and demonstrating higher burnup fuel—which is the most straightforward near-term means for significantly improving uranium utilization. Work will also be conducted on the development and demonstration of other promising methods for improving LWR uranium utilization.

Another important activity will be an Advanced Reactor Design Study for improved uranium utilization which will be initiated in fiscal year 1979. This study will be aimed at assessing the maximum attainable improvement in uranium utilization which can be achieved in a once-through LWR, employing both the backfittable improvements now under development and non-backfittable (plant design) changes.

The once-through improved LWR has a number of advantages. The proliferation resistance of its fuel cycle is acceptable if the enrichment facilities and spent fuel are adequately safeguarded. There is no material in any part of the fuel cycle that is readily usable for weapons. Less spent fuel will be generated in the improved LWR, resulting in reduced storage requirements and lower storage costs. The LWR is by far the most widely used reactor in the United States and abroad, and it has proven itself to be a safe, reliable, and economic source of power.

We have already stimulated a high degree of international interest in our program to improve the uranium efficiency of light water reactors. This has been done through presentations to international forums (the 10-nation, Norwegian-based Halden project group, INFCE Working Group 8, internationally attended American Nuclear Society technical meetings). We have been approached by several foreign and domestic groups seeking to establish multilateral international cooperative research and development projects in these technologies. Three other multilateral projects are under consideration, under British, Swedish and Danish leadership. The DOE intends to pursue vigorously each of these initiatives and to continue to stimulate foreign interest in improved uranium utilization.

Advanced Isotope Separation Technology.—The budget request for fiscal year 1980 for Advanced Isotope Separation Technology is $55.0 million in budget authority, compared to $54.2 million in fiscal year 1979.

This technology offers the prospect of significantly extending the uranium resource base available to the LWR. Gaseous diffusion enrichment plants have been operating in this country since the 1940's. Depleted tails containing about 0.2 to 0.3 percent U^{235} are a waste product of these plants. The Advanced Isotope Separation Technology, if successful, could economically improve extraction efficiency of U^{235} from ore and extract additional fissile material from the tails stockpile. This could increase the available energy from uranium ore by 20 percent.

Three processes are under development. They are the molecular process, the atomic vapor process, and the plasma separation process. The first two processes are based on the use of lasers and the third process uses ion cyclotron resonance. Progress in the development of these advanced separation processes continues to be encouraging. By the end of fiscal year 1981, we should have a sufficient data base to understand the science and the scaling issues of all three technologies. This knowledge will assist us to better evaluate the economic, environmental, and proliferation risks of the three technologies. In particular, the proliferation risk of AIST will be evaluated as the technology evolves.

National Uranium Resource Evaluation.—The fiscal year 1980 budget request for the National Uranium Resource Evaluation (NURE) program is $80.3 million in budget authority compared to $69.4 million in fiscal year 1979. This program is under the direction of the DOE Assistant Secretary for Resource Applications.

The NURE program is seeking to substantially improve knowledge of U.S. uranium resources, and, hopefully, higher resource estimates will result. This effort, which has been underway since 1974, is using a systematic approach to develop the field data needed to assess geologic favorability for the occurrence of uranium, which is the basis for resource appraisal. The primary focus of the NURE program has been on the lower-cost resources, that is, those producible at a cost of $50 per pound U_3O_8 or less. However, higher-cost materials, principally in the range of $50 to $100 per pound, are also being investigated.

LWR ALTERNATIVES

There are two fuel efficient alternatives to the LWR: the heavy water reactor (HWR) and the high temperature gas cooled reactor (HTGR). Canada is the supplier of commercial HWRs (CANDU) which are direct competitors to the LWR. With the LWR improvement program, it is possible to maintain the LWR's competitive position with the HWR in both fuel efficiency and economics so that it is not necessary to consider a U.S. HWR development program at this time.

The HTGR, on the other hand, has features that, if developed, could be complementary to the LWR and not necessarily in direct competition with it. The attractiveness of the HTGR stems from its potential capability to operate in arid regions abroad because of its high efficiency on the direct cycle. The HTGR operating on the once-through cycle with low enriched uranium (up to 20 percent) is considered to have superior proliferation resistant characteristics because of its fuel form and low fissile plutonium production rate.

Because of these potential advantages, the DOE has proposed a reorientation of its support program. With the direct cycle HTGR program, the DOE has revised its R. & D. efforts so that they are consistent with and supportive of the FRG program so as to maximize mutual benefits.

ADVANCED FUEL CYCLE SYSTEMS

Following the indefinite deferral of nuclear fuel reprocessing and recycle, the DOE fuel cycle R. & D. programs have been terminated except for LMFBR related R. & D. and NASAP studies of proliferation resistant concepts. NASAP activities are designed to examine systems which (1) utilize processes less efficient at fission product removal than the conventional purex (partially decontaminated fuel), (2) produce alternative fuel forms (i.e., coprocessing and/or spiking) using solvent extraction (purex) technology, and (3) examine more proliferation-proof methods of engineering key equipment and maintenance systems in critical reprocessing operations (proliferation-resistance engineering).

The LMFBR alternative fuel designs being studied include thorium/U^{233} cycles that prevent direct access to weapons grade materials and Pu fueled designs to be located in a few secure areas with thorium blankets that can be a source of denatured fuels to be used in reactors outside the secure areas. These denatured fueled reactors would be thermal converters with good fuel utilization to achieve a high ratio of denatured reactors to breeder reactors.

Preliminary NASAP assessments of other advanced concepts such as the Gaseous Core Reactor, Accelerator Breeder, Molten Salt Reactor and Fusion-Fission Hybrid indicate that they are not viable alternatives. They have one or more of the following drawbacks:

Too closely to enter the marketplace.

Pose difficult technological feasibility problems.

Require sufficiently long development time to negate their impact on the marketplace.

Offer, at best, only marginal proliferation resistance improvements.

Fiscal year 1980 funding for alternative LMFBR designs and technological improvements to fuel cycle facilities, which is included in the proposed budget for the LMFBR, is $27 million.

REDUCED ENRICHMENT FOR RESEARCH REACTORS

The Reduced-Enrichment Research and Test Reactor (RERTR) Program was established to develop and demonstrate reduced-enrichment uranium (REU) fuels in research and test reactors, to improve the proliferation-resistance of

related fuel cycles, and to facilitate international use of such reduced-enrichment fuels and thereby effect a reduction in U.S.-supplied highly enriched uranium (HEU) inventories abroad. The near-term objective of the program is to demonstrate the use of current technology to achieve the lowest feasible fuel enrichment. Fuel fabrication and core technology currently available in the United States and Europe permits enrichment reduction from 90 to 93 percent to below 20 percent in most reactors. Some high performance research reactors may still require 30 to 45 percent enriched fuel to avoid undue compromise in their performance in the near-term.

The long-term objective of the program is to develop high-uranium-density fuel technology to permit enrichment reductions to below 20 percent for essentially all existing research and test reactors and to provide satisfactory low enriched fuel for new reactors. An equally important long-term objective is to provide the technical support necessary to make the high-uranium-density fuels commercially available.

There are currently 141 operating research and test reactors, worldwide, representing 1689 MW of power. To maintain fuel supplies for these reactors, about 5 MT of HEU is currently in worldwide commerce. Using existing technology, approximately 90 reactors can utilize 20 percent enriched fuel and an additional 35 reactors can utilize 45 percent enriched fuel without significant performance impairment. Using advanced technology it is expected that all but about five reactors, representing about 800 MW, can be converted to 20 percent enriched fuel and those five should be able to convert to 45 percent (or less) enriched fuel. This means that worldwide commerce in HEU for research and test reactors could be cut at least in half, and depending on the possibility of converting the last five reactors, could potentially be reduced still further.

The budget for the RERTR Program is $4.1 million in fiscal year 1979 and $5.0 million (proposed) in fiscal year 1980.

This concludes the overview of DOE's principal programs in support of U.S. nonproliferation policy. I shall be happy to address myself to any questions you may have at this time. I should also note that answers to several specific questions posed by Committee are being submitted for the record with my testimony.

ANSWERS TO QUESTIONS SUBMITTED IN ADVANCE BY THE COMMITTEE

Question 1. What basic problems have to be resolved before low and medium enriched research reactor fuels can be licensed for export?

Answer. Most of the fuel for foreign research and test reactors is provided by foreign fuel fabricators, namely CERCA (France) and NUKEM (FRG). Export licenses are required for the export of the enriched uranium produced in the United States used to fabricate the fuel in foreign countries.

As far as licensing REU (reduced-enriched-uranium) fuels in the United States is concerned, the technical problems associated with licensing an REU fuel using conventional technology are largely resolved. For advanced technology, however, extensive fuel irradiation testing both transient and steady-state, will be required prior to licensing.

Question 2. How soon and by how much can we curb world traffic in highly enriched uranium as a result of the research reactor conversion program?

Answer. There are currently 141 operating research and test reactors, worldwide, representing 1689 MW of power. To maintain fuel supplies for these reactors, about 5 MT of HEU is currently in worldwide commerce. Using existing technology, approximately 90 reactors can utilize 20 percent enriched fuel and an additional 35 reactors can utilize 45 percent enriched fuel without significant performance impairment. Using advanced technology it is expected that all but five reactors, representing about 800 MW, can be converted to 20 percent enriched fuel and those five might be able to convert to 45 percent (or less) enriched fuel. Thus, if the technology is accepted, worldwide commerce in HEU for research and test reactors could be cut at least in half, and depending on the possibility of converting the last five reactors, could potentially be reduced still further.

Question 3. Is the conversion schedule relatively insensitive at this point to higher funding levels, or could additional funding speed up the conversion program significantly?

Answer. We do not have a conversion schedule, per se, since decisions to switch to REU fuel will be made by the reactor operators when the fuel is available. Our mission is technology development and demonstration in a manner which encourages adoption of the technology. Although the implementation schedule is constrained by time of developing and testing advanced fuels, the confidence of the reactor operators to use REU fuels will only come about as a result of the demonstration of REU in existing reactors. Our present program already envisions several demonstrations and we feel that these would be adequate. Additional funding could be used to perform more demonstrations, but if the planned demonstrations are indeed successful, as we hope they will be, then more demonstrations would not necessarily result in more confidence on the part of research reactor operators.

Question 4. In evaluating advanced reactor systems, does the Department take the Committee's "timely warning" criterion into account?

Answer. The assessment of advanced nuclear systems is being undertaken in the Department's Nonproliferation Alternative Systems Assessment Program (NASAP). The proliferation resistance of alternative nuclear systems is assessed by considering the activities that a potential proliferator would have to undertake to obtain nuclear explosives. Specifically addressed are the resources required, the detectability, and the approximate time intervals needed to carry out the activities required for a specified proliferation pathway including preparation, diversion and conversion. For existing nuclear systems, these times can be discussed in relation to existing IAEA safeguards systems and practices and the goals for timely warning. The detection time, which is the maximum time between diversion and detection by IAEA safeguards, can be taken as a parameter of timeliness. The detection time should correspond in order of magnitude to the conversion time, which is the minimum time required to transform fuel cycle material to the metal components of a nuclear explosive. Estimating these times requires judgment and involves numerous uncertainties. Assessing whether the time intervals estimated for proliferation pathways would yield timely warning under different scenario assumptions is also subject to uncertainty, even in the case of existing systems and the existing IAEA safeguards regime. Potential improvements to this regime are being identified and assessed for certain closed fuel cycles such as the LMFBR where technical information is fairly well developed. In order to assess the prospects of timely warning in the case of other advanced systems, it would be necessary to postulate the specific safeguards regime appropriate to such systems when they would be deployed. These systems, however, could not be deployed for many decades. The technical uncertainties surrounding such advanced systems are significant enough to preclude useful speculation at this time of the possible future safeguards regime that would be appropriate.

Question 5. How does the Department's work in the International High Temperature Gas Cooled Reactor (HTGR) program serve U.S. nonproliferation aims?

Answer. In fiscal year 1980, the Department is proposing to shift its support away from the HTGR steam cycle and reorient the program to emphasize the development of the Direct Cycle HTGR and the Very High Temperature Reactor (VHTR) for high temperature industrial process heat applications. There appeared to be insufficient national incentives and private sector investment commitment to warrant continued government support for the HTGR steam cycle. On the other hand, the Direct Cycle HTGR and the VHTR have the potential for exploiting the unique high temperature capability of gas cooled thermal reactors such as economic dry cooling, application to a new nuclear market (high temperature process heat), high thermal efficiencies, etc. In 1977, the Federal Republic of Germany (FRG) reoriented its program to emphasize the Direct Cycle and VHTR reactors. If the Department proposal to reorient the HTGR program is accepted by Congress as a part of the fiscal year 1980 budget process, then the United States and Federal Republic of Germany programs will be focused on the same concepts. In fiscal year 1980, the HTGR program will be coordinated with the FRG program and they will be mutually supportive consistent with the existing international Gas Cooled Reactor Agreement. Particular emphasis will be given by the United States to low and medium enriched fuel development in an effort to further nonproliferation initiatives.

It is our understanding that a subcommittee of the House Science and Technology Committee has recommended $4.4 million for HTGR reprocessing. We feel that there is a basic inconsistency in the use of the HTGR program to promote our nonproliferation objectives if we are required to develop reprocessing for the HTGR. The Administration remains opposed to the continuation of research on recycle for the HTGR.

Question 6. What can be done to enhance the international interest in and acceptance of U.S. light water reactor efficiency work?

Answer. We have already stimulated a high degree of international interest in our program to improve the uranium efficiency of light water reactors. This has been done through presentations of our program in international forums (the 10-nation, Norwegian-based Halden project group, INFCE Working Group 8, internationally attended American Nuclear Society technical meetings) and in numerous discussions with foreign visitors in the U.S. We have been approached by several foreign and domestic groups seeking to establish multilateral international cooperative research and development projects in technologies which will contribute to improved uranium efficiency in LWRs. The first of these projects, to investigate fission has release from high burnup fuel, is just getting started at Battelle-Northwest Laboratories in Richland, Washington. Three other multilateral projects are known to be under consideration, under British, Swedish and Danish leadership. Very preliminary discussions for information exchange and/or bilateral cooperative research projects have taken place with key individuals of four nations. The DOE intends to pursue vigorously each of these leads and to continue to stimulate foreign interest in improved uranium utilization through presentations and personal contacts.

Chairman ZABLOCKI. Mr. Pickering.

STATEMENT OF HON. THOMAS R. PICKERING, ASSISTANT SECRETARY, BUREAU OF OCEANS AND INTERNATIONAL ENVIRONMENTAL AND SCIENTIFIC AFFAIRS, DEPARTMENT OF STATE

Mr. PICKERING. Thank you, Mr. Chairman, and Chairman Bingham.

Chairman ZABLOCKI. Mr. Pickering is the Assistant Secretary for the Bureau of Oceans and International Environmental and Scientific Affairs, Department of State.

I failed to identify Mr. John M. Deutch, but he needs no introduction. He is the Director of the Office of Energy Research at the Department of Energy. And Mr. Van Doren is the Assistant Director of Nonproliferation at the Bureau of Arms Control and Disarmament Agency. So we have the Department of Energy, the Department of State, and the Arms Control and Disarmament Agency. I am sure there is close coordination between all of you, the three agencies.

Mr. Pickering.

Mr. PICKERING. Let me begin by reemphasizing your remarks. There is indeed very close coordination between the three of us. You have already received a great deal of information from Dr. Deutch, and the detailed nature of the technical programs related to proliferation resistance of nuclear fuel cycles. I will try to make my comments brief and to address mainly the relations of these programs to our international fuel cycle evaluation and nonproliferation efforts. The nonproliferation effort is a cornerstone of our foreign policy and of our overall hopes to establish a stable and peaceful international climate.

Our nonproliferation effort consists of a wide array of political, institutional, and technical measures. It includes as a basic element support for the NPT, and is related to our efforts to reduce regional tensions and increase security of states. Today I understand the discussion will focus primarily on the technical programs related to the nuclear fuel cycle, but we should keep in mind that significance of these technical areas can extend into much broader areas of the overall effort. The technical efforts can have a major impact on success of political and institutional measures. I will give some examples of this sort of relationship.

INFCE was proposed as an open evaluation of various fuel cycle questions. It is not a negotiation, and no agreements as such will be reached. It is heavily dependent on technical and analytic input. In launching INFCE, we were asking the international community to reopen and reexamine modes of fuel cycle development, and assumptions as to factual background.

Mr. Deutch has reviewed some of the specific Department of Energy programs of interest. These include work on improved light water reactors, advanced and fast reactors, the NASAP program, and the work on reduced enrichment in research reactors. Development of spent fuel storage capacity and waste management are also relevant.

The Ad Hoc Interagency Group on Non-Proliferation has conducted detailed reviews, and has determined with DOE that its overall R. & D. strategy to promote improvements in the light-water reactor, LWR, for the near and medium term and to develop breeder and possibly other advanced technology with the most proliferation-resistant characteristics possible as insurance for the long term is technically and economically sound. We believe it is the appropriate strategy, which takes into account both energy security and nonproliferation concerns and objectives.

INFCE

The DOE, as Dr. Deutch has pointed out, is the major source of U.S. technical participation in INFCE. The State Department effort on INFCE is led by Ambassador Gerard Smith, who is the overall policy leader for U.S. participation. The operational direction of the U.S. INFCE effort is carried out by a management committee, chaired by Ambassador Smith's deputy, and which includes DOE, State, and ACDA members. My Bureau for the State Department provides the major staff involvement. Approximately five Department officials devote a large fraction of their efforts to INFCE and closely related nonproliferation matters. In addition, funding for attendance of some key U.S. participants at INFCE meetings is provided by the International Organizations Bureau.

Of central importance in INFCE is to provide a balanced analytic product which can be useful to national decisionmakers as they consider the future role of nuclear power and plan fuel cycle activities. In doing this, we hope to convey the point that national situations differ, and that it need not be assumed that all countries should move immediately into reprocessing and breeder development programs. We also hope to provide a balanced indication of the economic, nonproliferation, and resource implications of thermal recycle. We feel these implications will lead to the conclusion that thermal recycle is not advantageous, either from an economic or nonproliferation point of view. In providing U.S. contributions to such an analytic and factual base, several U.S. programs are of particular importance. These include work on international uranium resource evaluation, development of means to improve uranium utilization in present thermal

reactors and development and implementation of geologic disposal of spent fuel, should that be judged desirable. All of these inputs contribute to a realistic understanding of fuel cycle options available, over time, to countries now in the early stages of utilization of nuclear power.

BASIS FOR COOPERATION

I should also mention U.S. programs on breeder technology and on HTGR's which Dr. Deutch discussed in some detail. These programs provide the United States, internationally, with a basis for cooperation with similar programs in other countries. Such cooperation is both a vehicle for mutually advantageous exchanges, and a means for the United States to convey our work and views on proliferation resistance considerations; for example, avoidance in the fuel cycle of presence of HEU or pure plutonium. It also provides a joint involvement in energy programs and some consequent measure of influence on fuel cycle directions in other countries through this involvement. For example, we see the HTGR program with Germany as an example of mutual interest to the United States and several other advanced countries, and we see cooperation as one means of conveying our interest in avoidance of HEU fuel. INFCE has been, in addition to a vehicle for making our views known, a means for us to increase our understanding of the perceptions and assessments of other countries. We have now a clearer understanding of the major role some of these countries hope fast reactors will play in their energy future. A continuing U.S. involvement in this area, as provided by our own programs, is clearly desirable.

Establishing a satisfactory factual and analytic basis for nuclear program decisions in various countries also implies a proper perspective on the role of nuclear power itself in relation to other options. In that connection, our program of cooperative bilateral assessments of energy program options for developing countries is extremely valuable. We plan to engage in several bilateral assessments in the coming year. Both, our partners and we, gain understanding from such detailed reviews.

You have raised some specific questions, many of which have been addressed in Dr. Deutch's testimony. Some others will be addressed in mine.

With regard to criteria, specific, numerical criteria for quantities of material, times, or difficulties may be neither desirable nor acceptable to the international community. Such criteria might either be so tight as to effectively rule out some fuel cycle activities even in those cases where they may be clearly advantageous on resource and economic grounds, or alternatively, to be so loose as to be ineffective. However, I believe we are in fact achieving major progress in gaining international acceptance of some fundamental points related to criteria.

First and most generally, it is now widely accepted that proliferation implications must be a substantial consideration in making fuel cycle decisions. Second, it is commonly accepted that presence of weapons-usable material, either separated plutonium or highly enriched uranium, HEU, should be minimized or avoided to the greatest extent practicable. This is not a quantitative result, but it is clear and demonstrable progress.

In your letter, you specifically mentioned the question of timely detection. A closely related factor has been one major component of our own assessment of fuel cycles. We have proposed in the INFCE Technical Coordinating Committee general assessment factors to be used by the working groups. These factors, in their simplest description, are resources required, time required, and detectability of diversion activities based on the fuel cycle or facility in question. These factors are increasingly used in contributions of other states to INFCE. We will undoubtedly not agree on specific numbers, but we are gaining acceptance on the basis of such factors of the need to minimize presence of material which is readily convertible to weapons use or to provide some additional institutional measures to decrease risks of diversion.

I would like to conclude my statement here and thank you again for the opportunity to appear on behalf of the Department of State.

Chairman ZABLOCKI. Ambassador Pickering, your prepared statement will be made a part of this record.

[Mr. Pickering's prepared statement follows:]

PREPARED STATEMENT OF HON. THOMAS R. PICKERING, ASSISTANT SECRETARY, BUREAU OF OCEANS AND INTERNATIONAL ENVIRONMENTAL AND SCIENTIFIC AFFAIRS, DEPARTMENT OF STATE

Mr. Chairman, I appreciate the opportunity to address the important matters raised in your letter. You have received information from Dr. Deutch on the detailed nature of technical programs related to proliferation resistance of nuclear fuel cycles. Therefore, I will make my comments brief, and will address mainly the relation of these programs to our INFCE and non-proliferation efforts.

As the Committees are aware, the non-proliferation effort is a key element of U.S. foreign policy and of our overall hopes to establish a stable and peaceful international climate. Our non-proliferation effort consists of a wide array of political institutional, and technical measures. It includes as a basic element support for the NPT, and is related to our efforts to reduce regional tensions and increase security of states. Today we will discuss primarily the technical programs related to the nuclear fuel cycle, but we should keep in mind that significance of these technical areas can extend into much broader areas of the overall effort. The technical efforts can have a major impact on success of political and institutional measures. I will give some examples of this sort of relationship.

INFCE was proposed as an open evaluation of various fuel cycle questions. It is not a negotiation, and no agreements as such will be reached. It is heavily dependent on technical and analytic input. In launching INFCE, we were asking the international community to reopen and re-examine modes of fuel cycle development, and assumptions as to factual background. This is never an easy kind of thing to do. We were also accepting as a basic approach that ways must be found by which benefits of nuclear power are widely available, without unacceptable proliferation risks. This approach must be credible to assure the continued viability of the NPT. The United States bears the primary burden of proof that this proposition and the related technical questions are legitimate and answerable from an operational perspective. Thus, our related programs and support work are of considerable significance in supporting the evaluation and demonstrating the seriousness of our concern. Mr. Deutch has reviewed the specific DOE programs of interest; these include work on improved light water reactors, advanced and fast reactors, the NASAP program, and the work on reduced enrichment in research reactors. Development of spent fuel storage capacity and waste management are also relevant.

The Ad Hoc Interagency Group on Non-Proliferation has conducted detailed reviews, and has determined with DOE that its overall R. & D. strategy to promote improvements in the Light Water Reactor (LWR) for the near and medium term and to develop breeder and possibly other advanced technology with the most proliferation-resistant characteristics possible as insurance for the long term is technically and economically sound. We believe it is the appropriate strategy, which takes into account both energy security and non-proliferation concerns and objectives.

The DOE is the major source of U.S. technical participation in INFCE. Their effort has been major and of high quality. The State Department effort on INFCE is led by Ambassador Gerard Smith, who is the overall policy leader for U.S. participation. The operational direction of the U.S. INFCE effort is carried out by a Management Committee, chaired by Ambassador Smith's Deputy, and including DOE, State, and ACDA members. The OES Bureau, for the State Department, provides the major staff involvement. Approximately five Department officials devote a large fraction of their efforts to INFCE and closely related matters. In addition, funding for attendance of some key U.S. participants at INFCE meetings is provided by the International Organizations Bureau.

Of central importance in INFCE is to provide a balanced analytic product which can be useful to national decision makers as they consider the future role of nuclear power and plan fuel cycle activities. In doing this, we hope to convey the point that national situations differ, and that it need not be assumed that all countries should move immediately into reprocessing and breeder development programs. We also hope to provide a balanced indication of the economic, nonproliferation, and resource implications of thermal recycle. We feel these implications lead to the conclusion that thermal recycle is not advantageous. In providing U.S. contributions to such analytic and factual base, several U.S. programs are of particular importance. These include work on international uranium resource evaluation, development of means to improve uranium utilization in present thermal reactors and development and implementation of geologic disposal of spent fuel, should that be judged desirable. All of these inputs contribute to a realistic understanding of fuel cycle options available, over time, to countries now in the early stages of utilization of nuclear power.

I should also mention U.S. programs on breeder technology and on HTGRs. These programs provide the United States, internationally, with a basis for cooperation with similar programs in other countries. Such cooperation is both a vehicle for mutually advantageous exchanges, and a means for the United States to convey our work and views on proliferation resistance considerations (for example, avoidance in the fuel cycle of presence of HEU or pure plutonium). It also provides a joint involvement in energy programs and some consequent measure of influence on fuel cycle directions through this involvement. For example, we see the HTGR program as an option of mutual interest to the United States and several other advanced countries, and we see cooperation as one means of conveying our interest in avoidance of HEU fuel. INFCE has been, in addition to a vehicle for making our views known, a means for us to increase our understanding of the perceptions and assessments of other countries. We have now a clearer understanding of the major role some of these countries hope fast reactors will play in their energy future. A continuing U.S. involvement in this area, as provided by our own programs, is clearly desirable.

Establishing as good as possible factual and analytic basis for nuclear program decisions in various countries also implies a proper perspective on the role of nuclear power itself in relation to other options. In that connection, our program of cooperative bilateral assessments of energy program options for developing countries is extremely valuable. We plan to engage in several bilateral assessments in the coming year. Both our partners and we gain understanding from such detailed reviews.

You have raised some specific questions about the program to promote the use of lower enrichments in research reactors. You have also raised the question of progress on criteria to be used in evaluating proliferation resistance. I would like to relate these two points.

With regard to criteria, specific, numerical criteria for quantities of material, times, or difficulties may be neither desirable nor acceptable to the international community. Such criteria might either be so tight as to effectively rule out some fuel cycle activities even in those cases where they may be clearly advantageous on resource and economic grounds, or be so loose as to be ineffective. However, I believe we are in fact achieving major progress in gaining international acceptance of some fundamental points related to "criteria."

First and most generally, it is now widely accepted that proliferation implications must be a substantial consideration in fuel cycle decisions.

Second, it is commonly accepted that presence of weapons-usable material—either separated plutonium or highly enriched uranium (HEU)—should be minimized or avoided to the extent practicable.

This is not quantitative, but it is clear and demonstrable progress. These points had not been universally factored into fuel cycle decisions in the past. In order to sustain points such as this, we must make a credible case that these points are

not only desirable, but achievable without major negative impact on the uses of nuclear energy. In particular, our specific technical program on reduced enrichment in research reactors provides credibility that we can in fact avoid use of HEU. This program involves direct involvement with foreign fuel manufacturers and reactor operators to produce high density, lowered enrichment fuel, and to do operational tests on that fuel. Without this program we would have little chance to gain international acceptance of the need to avoid the use of HEU.

In your letter, you specifically mentioned the question of timely detection. A closely related factor has been one major component of our own assessment of fuel cycles. We have proposed, in the INFCE TCC, general assessment factors to be used by the Working Groups. These factors, in their simplest description, are: Resources required, time required, and detectability of diversion activities based on the fuel cycle or facility in question. These factors are increasingly used in contributions of other states to INFCE. We will undoubtedly not agree on specific numbers, but we are gaining acceptance, on the basis of such factors, of the need to minimize presence of material which is readily convertible to weapons use or to provide some additional institutional measures to decrease risks of diversion.

Chairman ZABLOCKI. Thank you very much.

Mr. Van Doren.

STATEMENT OF HON. CHARLES N. VAN DOREN, ASSISTANT DIRECTOR, BUREAU OF NON-PROLIFERATION, U.S. ARMY CONTROL AND DISARMAMENT AGENCY

Mr. VAN DOREN. Mr. Chairman, it is an honor for me to appear before this committee to discuss the role of alternative nuclear technologies in minimizing the proliferation dangers of the nuclear fuel cycle, and it is of particular pleasure to note that one of the greatest satisfactions of working on this subject in this current administration is that the three departments represented here are working in such close coordination, all in the same direction. That is a very constructive development.

Chairman ZABLOCKI. You don't have any obstruction from OMB in the process, do you?

Mr. VAN DOREN. Each of us has our own responsibilities. One of theirs is to keep the budget down.

ACDA considers the investigation of such alternatives to be an important aspect of U.S. domestic and international policy. In that regard, the United States has been successful in launching the international nuclear fuel cycle evaluation [INFCE], an unprecedented effort by the international community to investigate the nuclear fuel cycle and its proliferation implications. Domestically, a major effort to investigate alternative fuel cycles is currently being undertaken by the Department of Energy, under the nonproliferation alternative systems assessment program [NASAP]. Because of our strong interest, ACDA has also undertaken a modest but productive research program in this field.

But first, let me try to put this effort in perspective. The nuclear proliferation problem is an enormously complex one, to which no single approach is adequate. Local, political, and institutional approaches are of enormous importance, and every case has unique features. Thus we are under no illusion that any technical fix can solve the proliferation problem by itself.

We are also aware that there are routes to weapons capability that do not involve the commercial fuel cycle. But we are convinced that

minimizing the proliferation risks and the subnational threat involved in such cycles is essential to effective management of the overall problem.

Your committees have been in the forefront of those calling for efforts to insure that the United States will have timely warning of any diversion of nuclear materials well in advance of the time at which a State could transform the diverted material into a nuclear explosive device.

IMPROVING SAFEGUARDS

We have approached this problem from two angles: Improvement of safeguards against diversion and examination of ways of increasing proliferation resistance while still meeting nuclear energy needs. While the latter approach is the focus of these hearings, I would like to note the relevance of some current ACDA research to the first approach: Our largest safeguards research project is the design, development, and demonstration of a system to provide nearly instantaneous information to the International Atomic Energy Agency on the status of sensors at safeguard facilities. For example, it should enable the IAEA to check at any time on the status of seals placed on equipment or on stocks of nuclear material and thus materially help provide timely warning of any diversion.

Our efforts on the other approach have been focused on support for the international fuel cycle evaluation and the search for proliferation resistant fuel cycles.

Let me first describe briefly the resources that ACDA is devoting to this aspect of the problem, which is one of the major responsibilities of ACDA's Nonproliferation Bureau. My deputy, Dr. Rochlin, devotes virtually all of his time to this subject, and is a member of the U.S. INFCE Management Committee. The eight professionals in our Nuclear Energy Division, headed by Dr. Sheaks, are also engaged full-time in support of this effort, through in-house research and analysis, supervision of relevant external research projects, and active participation both in the eight INFCE working groups and in preparation of U.S. input to those groups. Our external research on this aspect of the problem, which complements that of the Department of Energy, is at a level of about $1 million per year, and we have also been able to call on outstanding consultants to aid in these studies.

Mr. Chairman, I respectfully request permission to insert in the record at this point a paper describing some ACDA initiatives related to nonproliferation assessments and providing more specific details as to the relevant portion of our external research program for fiscal years 1978, 1979, and 1980.

Mr. ZABLOCKI. Without objection, it is so ordered.

[The material referred to follows:]

SOME ACDA INITIATIVES RELATING TO NONPROLIFERATION ALTERNATIVE ASSESSMENT

[Supplied by the Arms Control and Disarmament Agency]

ACDA was among the first to identify the need to take due account of proliferation resistance in making nuclear fuel cycle decisions and the need for intensive examination of alternative fuel cycles.

Its first initiative was suggestion of the so-called "tandem" fuel cycle, designed to recover the residual fuel value from the spent fuel from light water reactors through its use—without reprocessing—in CANDU (i.e., heavy water) reactors. Further examination of this fuel cycle was later included in the Department of Energy NASAP program. While the prognosis for adoption of this particular system does not currently look bright, it catalyzed investigation of other ideas having similar goals.

A second area in which an ACDA initiative was subsequently pursued by the Department of Energy was the development of densified lower enriched substitutes for the weapons grade uranium currently being used in many research reactors. We also instituted work on the conversion of research reactors using natural uranium to the use of fuels of medium enrichment, to reduce the amount of plutonium produced.

More recently, we have initiated a number of studies of improved once-through systems—which conserve uranium resources (and enrichment) without reprocessing. One example is a contract study being done on a modification of the light water breeder reactor to permit enhanced fuel savings in a once-through mode. This modification would entail neither the use of highly enriched uranium nor the separation of plutonium. Similarly, we have sponsored projects on the use of uranium fuel of lower enrichments in high temperature gas cooled reactors, on which the United States plans to do cooperative studies with the Federal Republic of Germany.

ACDA has also initiated studies on the use of thorium as fuel in light water reactors—and as blanket material in fast breeder reactors, with a view to reducing the quantities of plutonium produced and fostering the use of denatured fuel (that could not be used for weapons without isotopic separation).

ACDA has also been in the forefront of those seeking to develop and apply proliferation resistance criteria for use in INFCE and in subsequent fuel cycle decisions; stimulated studies on by-product and low grade uranium resources; promoted comparative economic analysis of alternative fuel cycles; and begun investigation of incentives needed to prompt industry to adopt alternative fuel cycles.

With respect to fuel cycles involving heavy water reactors, ACDA initiated a study of how to safeguard heavy water production facilities.

Attached is more specific information on ACDA's external research program on alternative systems assessments. With three exceptions (fiscal year 1978 projects 1s, 3s and 6s) this list does not include our research on international safeguards techniques, instrumentation and implementation, which is addressed in other questions.

Fiscal year 1978 (including supplemental)

	Program amount
1. Evaluation of methods of improving fuel. 1A. Utilization for once-through fuel cycles	$66,850
2. A concept for optimizing thorium utilization in LWR's in a once-through fuel cycle	50,000
3. NASAP–INFCE summer study group	35,000
5. Study on unproliferation features of HTGR's and GCFR's	80,000
6. Alternative nuclear technologies	75,000
7. Cost analysis of proliferation	75,000
8. Economic and evaluation comparison of proliferation-related technologies	80,000
9. Nonproliferation verification of laser isotope separation	100,000
10. Utility/fuel vendor incentives for denatured fueled cores	40,000
12. A characterization of the international reactor deployment schedule	10,000
14. Depletion benchmark and irradiation performance evaluation for PWR thoria urania fuels	40,000
15. Low-grade uranium resources	50,000
17. Cost analysis of alternative breeder fuel cycles	70,000
18. An evaluation of the international safeguards for alternative nuclear fuel cycles	150,000
3s. Impact of proliferation resistant fuel forms on international safeguards	12,000
6s. The design of heavy water production plants to facilitate the application of international safeguards	170,000
Total	[illegible]

Fiscal year 1979

	Program amount
5. Improvements in once-through fuel cycles	$210, 000
6. Utility/fuel vendor incentives for denatured fuel cores phase II	65, 000
7. Thorium-based fuel cycles	170, 000
8. Economic and evaluative analysis for nonproliferation	362, 000
9. Quick response analysis that relate to breeder and once-through systems	113, 000
10. An analysis of international nuclear fuel cycle facility materials inventories	80, 000
11. Quick response studies on alternate nuclear fuel cycle	10, 000
12. International nuclear fuel cycle data and analysis	10, 000
Total	1, 020, 000

Fiscal year 1980 program

	Funding level
A. Improvements in once-through fuel cycles	$300, 000
B. Thorium fuel cycles	120, 000
C. Economics and proliferation resistance assessment	300, 000
D. Nuclear fuel cycle data and inventories	80, 000
Total	800, 000

Mr. VAN DOREN. Let me now briefly describe three lines of research in this field that we consider especially promising.

First, improved once-through cycles. The principal argument used for the recycle of reprocesed fuel in light-water reactors was that it would result in saving uranium and enrichment services. If similar fuel savings can be achieved without reprocessing, the cost of reprocessing and the incremental cost of mixed oxide fuel fabrication could be saved. And we would reap the significant nonproliferation advantage of avoiding the separation and widespread circulation of plutonium. Moreover, the longer we can rely on sufficient uranium, the longer we have to find the optimum solution to choosing the next generation of nuclear reactors.

Dr. Deutch has described the Department of Energy programs on improved uranium utilization, which we strongly support. The DOE research has developed near-term fuel improvements which can lead to uranium savings in the neighborhood of 15 percent. We have been investigating along with DOE possible longer term improvements in LWR fuel use that might result in an additional 25-percent saving. (I note that Dr. Deutch cites the figure 10 to 15 percent additional savings; I think we need a lot more work before we can fix percentages.) We are also examining modification of the light water breeder reactor design that could result in even more dramatic resource savings operating in a once-through mode.

While we respect the reasons why DOE has not concentrated its efforts on heavy water reactors, it must be recognized that a number of other countries have purchased such reactors. We have seen utility in exploring the possibilities of improvements in such reactors from proliferation-resistance and resource points of view. One such possibility would be the use of 1 percent enriched uranium in such reactors, which would greatly decrease their uranium requirements and make reprocessing for recycle in such reactors even less attractive than it now is. We have also done considerable work with Canada and the

International Atomic Energy Agency on improved safeguards for such reactors.

THORIUM-BASED FUEL CYCLES

The second of the three fuel cycle alternatives that I would like to address briefly is the thorium-based fuel cycle. The substitution of thorium for fertile isotope uranium-238 in nuclear reactors—both thermal and breeder reactors—could have nonproliferation advantages if combined with new institutional arrangements, such as secure multinational energy centers and enhanced IAEA safeguards. The primary advantage stems from the fact that the fissile material produced in the reactor is the isotope uranium-233 which can be denatured—mixed with U^{238}—and thus rendered unusable for nuclear weapons without isotopic separation. Such enrichment technologies are likely to remain beyond the capacity of terrorist and subnational groups forever, and beyond the capacity of many nations for decades. We consider this advantage significant even though it is not a complete technological fix. In the second place, the production of U^{233} is always associated with another uranium isotope, U^{232}, a contaminant whose decay involves very strong gamma radiation, which complicates the separation and handling of the fissile material. This is an industrial disadvantage of the thorium cycle, but a valuable nonproliferation advantage. Finally, although irradiation of "denatured uranium-thorium" fuels does not eliminate the production of plutonium, it does greatly reduce it.

We believe these denatured fuel cycles may be of particular international interest with respect to the development of symbolic relationships between breeder and thermal converter reactors. And we have been promoting this in INFCE. (We have not gotten widespread endorsement from the world community on this idea, but we have not given up hope of doing so.)

It is well known that thermal reactors using U^{233} as a fuel have significantly improved conversion ratios, approaching unity for some advanced converter types. Those national programs which include plans for thermal reactors could, at least for the balance of this century, benefit from reduced uranium fuel requirements.

The DOE is developing technical information on LWR's, HTGR's, and HWR's operating on the thorium fuel cycle and has a substantial program of fast reactor thorium utilization. ACDA research in this area includes examination of incentives for the development of advanced converters operating on the denatured thorium cycle and for the implementation of thorium blanketed breeders.

REDUCED ENRICHMENTS

The third and last development I would like to touch on today as a highlight is the use of reduced enrichments in research reactors, of which Dr. Deutch has already spoken.

There has already been major progress in the investigation, development, and commercialization of fuels of lower enrichment for research and test reactors. There are many such reactors operating worldwide which now operate on weapons-grade uranium. Such highly enriched uranium—HEU—poses potential nonproliferation and nu-

clear terrorist dangers at least as great, or greater, than those posed by plutonium. Although this problem can be mitigated by safeguards and enhanced physical security, with substantial quantities of HEU moving in international commerce, drastic measures were needed, including a search for alternatives to HEU.

In April of 1977 the President announced plans to minimize HEU uses by requiring, among other things, a careful economic and technical justification for its use, by minimizing HEU inventories, and by seeking to convert existing research reactors to the use of lower enrichments as quickly as possible.

CONVERTING TO LOWER ENRICHMENTS

The Department of Energy has an excellent program, which Dr. Deutch described, underway at the Argonne National Laboratory to develop and test new high-density fuels which will permit a significant reduction of enrichment level in both new and existing reactors. This program is receiving considerable international interest and cooperation at the IAEA, at INFCE, and at a special meeting hosted late last year by the United States at Argonne National Laboratory. We are obtaining the active cooperation of foreign governments and fuel fabricators.

In addition, the General Atomic Co., working in conjunction with the Department of Energy, has already developed and begun marketing alternative low-enriched—20-percent enrichment—fuels [LEU] for its line of TRIGA reactors, and has announced that it is discontinuing HEU fuel fabrication for research reactors. We also expect that European fuel fabricators will begin making LEU and reduced-enrichment fuels—45 percent—commercially available within the next 2 years. Even further reductions may be possible through the research program that Dr. Deutch described.

As a result of these efforts, we are confident a substantial number of the world's research reactors can be converted to lower enrichments within the next few years. We consider this to be very beneficial from a nonproliferation perspective, and a highly cost-effective and timely payoff of investigating alternative technologies.

Of course, encouraging countries to actually convert their reactors will require suitable incentives. The executive branch has proposed to the Congress that the United States provide incentives to countries which wish to convert research reactors to lower enrichments. Such countries will necessarily face additional costs, which could pose a problem, particularly for developing countries. To offset these costs, the executive branch has proposed a program that would provide uranium enriched to 20 percent primarily to developing countries, with preference given to NPT parties. A companion program would offset incremental costs of fuel fabrication and would be offered to countries using HEU fuel in research reactors with special emphasis on developing countries. By making these offers, we would be complementing our technical program of reduced enrichments with programs that would provide real incentives to countries which possess HEU fuel and research reactors using HEU to return such fuel and to convert the reactors to lower enrichments.

ACDA has been involved in all aspects of this effort to reduce the HEU problem.

Finally, Mr. Chairman, while it is still too early to judge how these developments may be reflected in the final outcome of INFCE, we believe they are making an important contribution to that study, and helping to focus the attention of other nations on more proliferation-resistant alternatives to the fuel cycles that were long assumed to be the inevitable next step in nuclear power development.

Thank you, Mr. Chairman.

LOCATION OF REACTORS

Chairman ZABLOCKI. Thank you, gentlemen.

One question, Dr. Deutch. I almost wanted to interrupt you when you were giving your statement. You mentioned that a number of countries have research reactors and that there are five such reactors that are likely to be particularly hard to convert to nuclear fuel of 20 percent or less enrichment of uranium-235. Where are these 5 reactors located?

Mr. DEUTCH. The five reactors include the BR-2 reactor in Belgium, the HFR reactor in Grenoble, France, and the advanced test reactor, the high flux irradiation reactor and the General Electric test reactor in the United States.

Chairman ZABLOCKI. What are the problems?

Mr. DEUTCH. Excuse me, Mr. Chairman. There is a reactor at Oak Ridge. There is a reactor at Idaho which we currently fuel with highly enriched uranium which would not be possible to convert.

Chairman ZABLOCKI. Why would it be impossible? What would make the conversion so difficult?

Mr. DEUTCH. My understanding is that the uses of this reactor are for particularly high neutron fluxes for particular experiments. The one in Idaho is used for the Naval reactor program, and we need to get those neutron fluxes. Let me present it for the record, Mr. Chairman.

Chairman ZABLOCKI. Are these in the area of research?

Mr. DEUTCH. Yes.

Chairman ZABLOCKI. Is this research worth the risk if they are so difficult——

Mr. DEUTCH. That judgment can always be considered at a later date if they are, but it is my instinct that with proper safeguards we could quite happily live with that number of reactors. We would have lowered the amount of highly enriched uranium in general commerce significantly by just limiting ourselves to 5 in contrast to the present 141. It would be quite an achievement, and I think it could be done. Perhaps we could make those five operate down to 45 percent enrichment which would be a step forward, maybe not get them all the way down to 20 percent.

Chairman ZABLOCKI. You can amplify that for the record, if you will.

Mr. DEUTCH. Yes, Mr. Chairman, I will be happy to do so.

[The following was subsequently provided:]

ENRICHMENT REDUCTIONS

[Supplied by the Department of Energy]

These reactors are very high performance reactors which have pushed existing fuel technology to the limit. The uranium densities currently approach what

Exhibit F

DOE/S-0010R1

U.S. Department of Energy

Second Annual Report on Nuclear Non-Proliferation

Supplement to Secretary's Annual Report to Congress

1980

Pacific Basin Spent Fuel Storage

During 1979, DOE participated in discussions with Japan concerning the concept of an interim spent nuclear fuel storage capacity for the Pacific Basin area and made some preliminary conceptual studies of the possibilities for establishing such a facility. No decision to build one is anticipated in the near future. A primary consideration in determining whether to proceed with the effort will be the results of thorough study and evaluation of health, safety, environmental, political, social, and cultural factors. The relevant committees of Congress were informed during the year of the status of the discussions with Japan and the nature and extent of the preliminary studies. Those committees will be kept informed of further developments.

With respect to cooperation with India, DOE provided technical advice toward the reracking of the spent fuel storage pools of the Tarapur reactors. The Department of State arranged for the loan of equipment and the provision of consultant services for pool cleanout.

Views and Recommendations

The United States should continue actions to promote international cooperation in the storage of spent power reactor fuel. The Nation should also establish, as soon as feasible, a domestic program of away-from-reactor spent fuel storage with provision to accept limited quantities of foreign spent power reactor fuel when such action advances U.S. non-proliferation interests. This was proposed by the President in October 1977 and reaffirmed in February 1980.

Development of Proliferation-Resistant Fuel Cycle Technologies

Introduction

As part of its non-proliferation responsibilities, DOE has been examining a variety of fuel cycles and nuclear systems to determine if the proliferation risks of existing technologies might be reduced. The results of the Non-Proliferation Alternative Systems Assessment Program were published in June 1980.[2]

Reduction of Enrichment in Fuel for Research Reactors

In April 1977, the Administration decided to work toward minimizing the use and distribution of high-enriched uranium fuel. One element in the policy, a program known as the Reduced-Enrichment Research and Test Reactor Program (RERTR), is aimed at developing and encouraging the use of reduced-enrichment fuel in research reactors on a worldwide basis when such changes do not diminish reactor performance and do not affect safety standards.

Progress was made in several areas of the RERTR Program during 1979. Foreign and domestic manufacturers were encouraged to develop and apply the technology required to produce reduced-enrichment fuels. General Atomic (United States), NUKEM (Federal Republic of Germany), CERCA (France), CNEA (Argentina), and possibly others are participating in fuel fabrication and in the development of the required technology. France is also conducting a reduced-enrichment fuel program for research reactors. During INFCE, Germany announced the start of a similar program. Extensive cooperation exists between the United States and various country programs, and international meetings are held periodically to discuss the progress in the development of research reactor fuel utilizing uranium of lower enrichments. Cooperation between domestic and foreign programs is being discussed. As a result of a U.S. initiative, beginning in 1980 international technical expert teams and fellowships for personnel are offered through the IAEA. They are structured to provide the staff and technical resources to those research reactor operators who lack sufficient capabilities to undertake independent conversion programs.

It is expected that prototype assemblies containing low-enriched uranium fuel (enriched to less than 20 percent U-235) for TRIGA reactors will be

[2] U.S. Department of Energy, *Nuclear Proliferation and Civilian Nuclear Power: Report of the Non-Proliferation Alternative Systems Assessment Program*, Executive Summary plus nine volumes, Washington, D.C., June 1980: DOE/HE-0001/1-9.

introduced into the 14-megawatt electric TRIGA research reactor in Romania under the low-enriched uranium fuel procurement assistance program announced by the United States at the 1978 U.N. General Assembly Special Session on Disarmament and discussed on page 3 of this report. The United States was the host of an information meeting on reduced-enrichment fuel for research reactors in November 1980.

Views and Recommendations

General international acceptance of the use of low-enriched fuels in research reactors in place of high-enriched fuels would represent a highly valuable contribution to reducing the risk of proliferation associated with nuclear research and development activities. INFCE has endorsed the use of such fuels. As indicated above, a number of nations are taking steps to develop and to use such fuels in their research reactors. In the light of these facts, the United States should continue programs designed to encourage the wider use of reduced-enrichment research reactor fuels.

The International Nuclear Fuel Cycle Evaluation

At an organizing conference in Washington, D.C., in October 1977, the United States joined more than 50 nations and 4 international organizations in one of the most comprehensive examinations of the nuclear fuel cycle yet undertaken. This International Nuclear Fuel Cycle Evaluation continued for more than 2 years, concluding its work at its second and final plenary session, held in Vienna, February 25-28, 1980.

A final evaluation of INFCE will not be possible for some time. This is because much of its value will depend on the degree to which participating states consider INFCE's findings in relation to their respective nuclear programs and the degree to which pending issues become the subject of technical studies and negotiations leading to new arrangements for the fuel cycle.

As reported in 1978, INFCE functioned through eight working groups covering all aspects of the fuel cycle. The reports of these eight groups, together with a summary and overview prepared by the Technical Coordinating Committee, were referred to governments for use in planning and executing their respective nuclear programs. The conference communique stated that the findings of INFCE have strengthened the view that:

1. Nuclear energy is expected to increase its role in meeting world energy needs and should be widely available for that purpose.

2. Effective measures can and should be taken to meet the specific needs of developing countries in the peaceful uses of nuclear energy.

3. Effective measures can and should be taken to minimize the danger of proliferation of nuclear weapons without jeopardizing energy supplies or the development of nuclear energy for peaceful purposes.

From the U.S. perspective, INFCE was successful in many respects. All participants now more widely share the view that substantial risks are associated with the use of weapons-usable materials in the fuel cycle and the technology required to produce them. The collective acceptance of this premise should now help to steer nuclear power in safer directions.

DOE will be involved in the continuing efforts of the U.S. Government to implement new protective measures associated with the next steps in the nuclear fuel cycle, including technical changes, institutional arrangements, and improved safeguards.

Views and Recommendations

Now that INFCE has concluded, the United States should be prepared to pursue areas of agreement through appropriate post-INFCE fora leading to new institutional arrangements and technical decisions regarding the fuel cycle. DOE should also be prepared to support these efforts with an appropriate capability to undertake studies, provide for technical exchange, and implement such steps as spent fuel storage cooperation.

INFCE - Int'l Nuclear Fuel Cycle Evaluation

Exhibit G

DOE/PE-0014/R1

First Annual Report on Nuclear Non-Proliferation

Supplement to
Annual Report to Congress

U.S. Department of Energy
Office of Policy and Evaluation
Washington, D.C. 20585

a fast, once-through, breeder reactor, it offers a number of proliferation-resistant features that include no recovery or use of plutonium except on an in situ basis. The reactor concept offers an increase in uranium fuel utilization by approximately a factor of 15 over the conventional LWR without reprocessing and recycle. Disadvantages are very long fuel residence times and the need for reactor materials to withstand very high burnups—materials not yet available today. The FMRS concept, which would basically draw upon existing LMFBR or GCFR technology, is currently being studied by DOE.

This discussion has identified some of the alternative systems being considered in NASAP and other parts of DOE, and has tentatively indicated some of the features that may affect proliferation resistance. Further understanding of the effects of these features and identification of others will be an important function of the studies being conducted in NASAP.

Research Reactors

Numerous research and test reactors now in operation or planned were designed to utilize 90 to 93 percent enriched uranium to maximize flux performance per unit power and/or to minimize fuel cycle costs. Fabrication, transport, and storage of fuel for these reactors, particularly in the un-irradiated form, are of concern from a proliferation point of view. The larger fuel inventories associated with high-power test reactors increase the potential consequences of diversion. Elimination or substantial reduction of the trade in highly-enriched fuel elements for research and test reactors by substitution of reduced-enrichment fuel elements would lower the potential for using research and test reactor fuel as a source of material for nuclear explosives.

A program is underway in the United States to make feasible the fueling of most research and test reactors with uranium of less than 20 percent enrichment while maintaining the reactor performance. A small number of high-power, high-performance reactors needed for important work that cannot be reasonably accomplished in reactors with lower performance might have to continue to use high-enriched uranium. It is recognized, however, that for research and test reactors of power greater than a few megawatts, fuel technology does not currently exist that would permit enrichment reductions to below 20 percent without severe reactor performance reductions (flux per unit power), expensive reactor modifications, and/or fuel cycle cost increases relative to highly enriched designs using 90 to 93 percent enriched uranium. The program now beginning in the United States is designed to develop the necessary fuel technology. Several years of work will be needed.

Currently proven fuel technology is capable of accommodating enrichment reductions to the 45 percent range (from 90 to 93 percent) without significant performance degradation or fuel cycle cost increase for many reactors in the 1 to 50 megawatt range (some reactors can be converted to less than 20 percent enrichment). Accordingly, as an immediate interim step, the United States is proposing to convert existing research and test reactors (and new designs) from the use of highly enriched fuel to the use of either 45 percent enriched fuel or 20 percent enriched fuel wherever this can be done without unacceptable reactor performance degradation. It appears this can be achieved without significant cost increase.

Proliferation Resistance

HEU Research Reactors: The removal of HEU from a research reactor to obtain sufficient material to build a nuclear explosive would require removal on a scale comparable to the annual fuel element requirement for a typical large research reactor. For instance, a 20 megawatt (thermal) research reactor may have about 200 grams of HEU in each fuel element. About sixty fuel elements are needed as replacements each year. For this example, more than an annual supply of fuel elements would have to be diverted to build a nuclear explosive. However, the fabrication of fuel elements for a given research reactor is normally performed on a special order basis and may involve considerable leadtimes. Thus, in the absence of measures to minimize HEU inventories, typical procurements of fresh fuel elements would otherwise be available and stored at the reactor site. Significantly large quantities of HEU are also present at the fuel fabrication facilities.

Moreover, large quantities of irradiated HEU can build up at research reactors, even ones of substantially lower power (e.g. 1 to 5 megawatts (thermal)).

LEU or Natural Uranium Reactors: Natural uranium-fueled research reactors produce plutonium at the approximate rate of 1 gram per megawatt (thermal) per day of operation. A typical natural uranium fueled 20 MWt research reactor would therefore produce about 5 kilograms of plutonium per year. The amount of plutonium produced is reduced as the enrichment level is increased. A 20 MWt research reactor using 10 to 20 percent enriched uranium would generate about 0.5 kilograms of plutonium per year.

The proliferation resistance of spent fuel from research reactors would be similar to that from nuclear powerplants with the following exceptions:

1. The amount of radioactivity from research reactor spent fuel can be as small as one fiftieth that of fuel from a commercial power reactor, so shielding problems may be less difficult to deal with.
2. There are several different chemical forms that are typically used for research reactor fuel elements, so that the steps involved in the chemical reprocessing would be altered.

Efforts to Improve the Proliferation Resistance of Research Reactors

The U.S. development program for enrichment reduction in research and test reactor designs currently using 90 to 93 percent enriched uranium is based on the practical criterion that enrichment reduction should not cause significant flux performance (flux per unit power) or burnup performance degradation relative to the unmodified reactor design. To first order, this implies the requirement that the U-235 density in the reduced-enrichment case be the same as the U-235 density in the 90 to 93 percent enriched case. This can be accomplished by substitution of higher uranium density fuel technology for currently used fuel technology. Enrichment reduction potential is set in proportion to the available uranium density increase. It is recognized that, for research and test reactors of power greater than a few megawatts, fuel technology does not currently exist that would permit enrichment reductions to below 20 percent. As already indicated, a program is now beginning in the United States to develop the necessary fuel technology. The program is expected to last for several years.

Improved international safeguards and a more universal commitment to full scope safeguards would also be important for increasing the proliferation resistance of research reactors. Safeguards procedures need to accommodate the necessary flexibility of research reactor operations.

A long-term goal would be the achievement of a level of enrichment of between 3 and 20 percent. Enrichments in this range would maximize research reactor proliferation resistance. Increasing the enrichment of natural uranium research reactors to about 3 percent would substantially reduce their plutonium production and hence the availability of weapons-usable material in the spent fuel. Efforts to make existing technologies available on a commercial basis could make a significant contribution toward meeting this goal.

Exhibit H

DOE/NE-0001

Nuclear Proliferation and Civilian Nuclear Power

Report of the Nonproliferation Alternative Systems Assessment Program

Executive Summary

U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Washington, D.C. 20545

June 1980

TABLE 1. RESEARCH AND DEVELOPMENT PROGRAM RECOMMENDATIONS

PROGRAM	TARGET
Light-Water Reactor Fuel-Utilization Improvements:	
• High burnup, and operational and fuel-management changes	Commercial implementation by 1990
• Additional high burnup and other fuel-design changes	Commercial implementation by 2000
• Longer-term nonretrofittable improvements	Identify initial attractive candidates by mid-1980; commercial capability after 2000
Reduced Enrichment of Research Reactor Fuel	Demonstrate 20-45% enriched fuel by 1982; demonstrate <20% enriched fuel by 1984
Liquid-Metal Fast Breeder Reactor	Continue development so that it could be commercially available if and when needed (possibly 2010-2020)
Advanced Isotope Separation	Demonstrate technical and economic performance of a process by 1990-1995
Light-Water Breeder Reactor	Continue development and complete proof of breeding demonstration in Shippingport Atomic Power Station by 1985 or 1986
Proliferation-Resistance Engineering of Reprocessing	Demonstrate in breeder program pilot fuel-cycle facilities
High-Temperature Gas-Cooled Reactor	Assess unique markets, such as those for process heat and usability at water-poor sites
Fast Mixed-Spectrum Reactor	Investigate high-burnup fuel technology
National Uranium Resource Evaluation	Complete program by 1985. Continue research and development in discovery and extraction methods
Technology Support for IAEA Safeguards on: • Enrichment plants • Interim spent-fuel storage • Spent-fuel disposal repository • Reprocessing plants • Plutonium storage • Mixed-oxide fabrication plants • Transportation	Continued improvement in surveillance, containment, and material accountancy

Exhibit I

FEDERAL REGISTER (EXPORT/IMPORT)—Continued

Name of applicant, date of application, date received, and application number	Material type	Material in kilograms: Total element	Material in kilograms: Total isotope	End-use	Country of destination
Mitsubishi Int'l Corp., July 21, 1982, Aug. 9, 1982, XSNMO1980.	3.25 pct enriched uranium	16,788	643	Reload fuel for Ohi-2	Japan.
General Electric Co., Aug. 5, 1982, Aug. 9, 1982, XSNMO0463(06).	3.85 pct enriched uranium	[1]3,734	[1]82	Increase quantity of material for Caorso reactor, extend date, add intermediate consignee-fuel for Caorso.	Italy.
Total		126,586	8,066		

[1]Additional.

[FR Doc. 82-23045 Filed 8-23-82; 8:45 am]
BILLING CODE 7590-01-M

Use of High-Enriched Uranium (HEU) in Research Reactors; Policy Statement

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Statement of policy.

SUMMARY: The Nuclear Regulatory Commission (NRC) has licensing responsibility for domestic use and for export abroad of Special Nuclear Material, including High-Enriched Uranium (HEU), and is interested in reducing, to the maximum extent possible, the use of HEU in domestic and foreign research reactors. The NRC is pleased to note that the current U.S. Administration continues to support the Reduced Enrichment for Research and Test Reactors program and that to date the U.S. Congress has approved adequate funding for this program. In this connection, the NRC has prepared the following policy statement.

FOR FURTHER INFORMATION CONTACT: James V. Zimmerman, Assistant Director, Office of International Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555, (301) 492-7866.

SUPPLEMENTARY INFORMATION:

In the 1950's the U.S. entered into several short-term agreements for cooperation (5–10 years) allowing for the export of research reactors and fuel under the "Atoms for Peace" program. In subsequent years the U.S. has been a major supplier of high-enriched uranium (HEU) for use abroad, primarily in research and test reactors. Such reactors produce radioisotopes for use in such areas as medicine, agriculture, desalination, research in biological effects of radiation, etc. Materials test reactors are also used to train future operators of commercial power reactors and to test new materials and fuels.

In the mid 1970's, particularly following India's detonation of a nuclear explosive device in 1974, nuclear proliferation concerns began to increase. Expanded efforts were undertaken to prevent nuclear power programs from being exploited to produce nuclear weapons. Particular concerns were expressed with respect to the proliferation risks associated with inventories of HEU for research and test reactors abroad. The widespread use of HEU fuel, which involved a large number of domestic and international fuel shipments, increases the risks of proliferation through theft or diversion of this material. In contrast to HEU, the use of fuel with lower enrichments reduces proliferation risks.

In an effort to allay concerns of proliferation risks, efforts were made to reduce HEU inventories, on the assumption that any reduction in the potential for access to these inventories would constitute a reduction in the proliferation risk. These concerns eventually led to the establishment of the reduced enrichment for research and test reactors (RERTR) program. This program was established to develop and demonstrate the technology that will facilitate the use of reduced-enrichment uranium fuels in research and test reactors. If successful, this could lead to a significant reduction of HEU inventories abroad, and thereby increase the proliferation resistance of related fuel cycles.

The objective of the RERTR program is to develop research and test reactor fuels which will allow substitution of uranium of low enrichment (LEU, less than 20%) for HEU and which will not significantly affect reactor performance characteristics or fuel cycle costs. On an interim basis, some reactors may utilize intermediate enrichment fuels (45%), while the LEU fuel development program is in progress. It should be noted, however, that no U.S. effort will be made to develop fuels with enrichments significantly below 20%, because of the increasing magnitude of plutonium production in fuels with very low or no enrichment.

To date, DOE has initiated a development and test program managed by the Argonne National Laboratory (ANL) to prove the feasibility of the new lower enrichment fuels. Many foreign countries are cooperating with the U.S. in this effort, and, within the past year, NRC has issued several export licenses for reduced-enrichment uranium to be fabricated into test elements for foreign and domestic research reactors.

Assuming RERTR program success, most of the performance testing of LEU aluminide and oxide fuels with high uranium densities for use in plate-type reactors will be completed by the end of 1984. The irradiation of pin-type zirconium hydride fuel with high uranium density for use in Triga-type, and possibly plate-type, reactors will be completed in 1983. Assuming licensing approvals, these fuels could then enter into full scale use in appropriate reactors. Silicide fuels with very high uranium densities are also being developed and tested by the RERTR program. These fuels may be needed for conversion of high power reactors.

As part of the overall RERTR program, Argonne conducts for DOE a technical and economic evaluation of each significant HEU export license application including the potential of the reactor for conversion to reduced-enrichment fuel within the planned availabilities of appropriate reduced-enrichment fuels. Nearly all potential conversion candidates have been evaluated. Technical conversion schedules are being planned by reactor operators based on demonstration and licensability of the fuel. Based on the technical and economic evaluation by ANL, a coordinated Executive Branch recommendation on the license application is developed by the Department of State and is submitted to the NRC.

The objectives of the RERTR program have been fully supported by NRC since its inception. The Commission has also utilized Argonne's analyses in support of its reviews of proposed interim exports of HEU, particularly with respect to determining the dates when conversion to lower-enriched fuels can be anticipated. The Commission is pleased to note that the current Administration continues to support the RERTR program and that Congress has approved adequate funding for the program.

The Commission also notes that several types of LEU fuel are currently being tested in DOE's RERTR program. As soon as all the necessary tests are completed, the Commission is prepared to act expeditiously to review the use of the new fuel in domestic research and test reactors licensed by NRC.

With respect to future export license applications for HEU, bearing in mind the Commission's responsibility to make an overall finding that each export would not be inimical to the common defense and security of the U.S., the Commission intends to continue its current practice of careful scrutiny to verify that additional interim HEU exports are justified. The Commission plans to continue to monitor the progress of the RERTR program so that it can understand what would be appropriate conversion schedules, and to encourage that actions be taken to eliminate U.S.-supplied inventories of HEU to the maximum degree possible.

The Commission notes that U.S. research reactor operators have shown little interest in converting to lower enrichment fuel. As part of a policy to strongly encourage conversion by foreign operators, the Commission will take steps [1] to encourage similar action by U.S. research reactor operators.

Dated at Washington, D.C. this 17th day of August, 1982.

For the Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 82-23051 Filed 8-23-82; 8:45 am]

BILLING CODE 7590-01-M

Abnormal Occurrence Report; Section 208 Report Submitted To the Congress

Notice is hereby given that pursuant to the requirements of Section 208 of the Energy Reorganization Act of 1974, as amended, the Nuclear Regulatory Commission (NRC) has published and issued the periodic report to Congress on abnormal occurrences (NUREG–0090, Vol. 5, No. 1).

Under the Energy Reorganization Act of 1974, which created the NRC, an abnormal occurrence is defined as "an unscheduled incident or event which the Commission (NRC) determines is significant from the standpoint of public health or safety." The NRC has made a determination, based on criteria published in the Federal Register (42 FR 10950) on February 24, 1977, that events involving an actual loss or significant reduction in the degree of protection against radioactive properties of source, special nuclear, and byproduct materials are abnormal occurrences.

This report to Congress is for the first calendar quarter of 1982. The report identifies the occurrences or events that the Commission determined to be significant and reportable; the remedial actions that were undertaken are also described. The report states that there were four abnormal occurrences at the nuclear power plants licensed to operate. The first involved diesel generator engine cooling system failures. The second involved pressure transients during shutdown. The third involved major deficiencies in management controls. The fourth involved a steam generator tube rupture. There were no abnormal occurrences for the other NRC licensees during the report period. The Agreement States reported no abnormal occurrences to the NRC.

The report to Congress also contains information updating some previously reported abnormal occurrences.

Interested persons may review the report at the NRC's Public Document Room, 1717 H Street, NW, Washington D.C. or at any of the nuclear power plant Local Public Document Rooms throughout the country. Single copies of the report, designated NUREG–0090, Vol. 5, No. 1, may be purchased from the National Technical Information Service, Springfield, Virginia 22161.

A year's subscription to the NUREG–0090 series publication, which consists of four issues, is available from the NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Microfiche of single copies of the publication are also available from this source.

Dated at Washington, D.C. this 16th day of August 1982.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 82-23053 Filed 8-23-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50–373]

Commonwealth Edison Co.; Issuance of Amendment to Facility Operating License

On April 17, 1982, the U.S. Nuclear Regulatory Commission (the Commission) issued Facility Operating License No. NPF–11, to Commonwealth Edison Company (licensee) authorizing operation of the La Salle County Station, Unit 1 (the facility), at reactor core power levels not in excess of 166 megawatts thermal (5 percent power) in accordance with the provisions of the license, the Technical Specifications and the Environmental Protection Plan.

The Commission has now issued Amendment No. 4 to Facility Operating License No. NPF–11, which authorizes operation of the La Salle County Station, Unit 1, at reactor core power levels not in excess of 3323 megawatts thermal (100 percent power) in accordance with the provisions of the amended license. In addition, the Amendment makes administrative modifications dealing with omissions, an addition and changes in the areas of exemption, reporting to the Commission, and completion date of equipment qualification; requires confirmation of vacuum breakers to withstand pool swell forces; and a license condition regarding HVAC systems with respect to operation above 5% and 50% power.

La Salle County Station, Unit 1 is a boiling water nuclear reactor located in Brookfield Township, La Salle County, Illinois. The amendment is effective as of the date of issuance.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the amended license. Prior public notice of the overall action involving the proposed issuance of an operating license was published in the Federal Register on June 9, 1977 (42 FR 29576–29577). The increase in power level authorized by this Amendment is encompassed by that prior public notice. Prior public notice of the administrative changes authorized by this Amendment was not required since these changes do not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impacts other than those evaluated in the Final Environmental Statement, its Addendum, and assessment of the effect 40 year license from issuance of this amendment since the activity authorized by the license is encompassed by the overall action evaluated in the Final Environmental Statement, its Addendum, and assessment of license duration. Further, with respect to the administrative changes in the Amendment, the Commission has determined that the issuance of this Amendment will not result in any

[1] Because the "steps" referred to in the above sentence have not been detailed or discussed, Commissioner Roberts does not agree to the sentence since it implies that a specific course of action will be followed by the NRC.

Exhibit J

RULEMAKING ISSUE
(Affirmation)

June 12, 1981

SECY-81-376

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: PHYSICAL SECURITY REQUIREMENTS FOR NONPOWER REACTOR LICENSEES POSSESSING A FORMULA QUANTITY OF SSNM

Purpose: To provide the Commissioners with (1) a status report on the 22 nonpower reactor licensees listed in SECY 79-187B; (2) a resolution of the issues listed in SECY 79-187C; (3) a discussion of alternative physical security requirements for nonpower reactors possessing a formula quantity or greater of SSNM; and (4) a recommendation on the preferred alternative.

Discussion:

Background

On July 24, 1979, the Commission approved a recommendation that nonpower reactor (NPR) licensees be deferred from implementing the requirements of the Safeguards Upgrade Rule, and that in the interim new Category II (§73.67) physical protection requirements as well as previous existing requirements (§73.60) be applied to nonpower reactor licensees who possess formula quantities of SSNM. The interim requirements were to continue in force until certain nonpower reactor issues were resolved and a determination was made on what physical protection requirements are actually needed at these particular nonpower reactor facilities, given the unique type, form, and enrichment levels of the reactor fuel. The Commission asked the staff for an interim status report in 120 days which would give a more definitive explanation of the nonpower

Contact:
C. K. Nulsen, SGRI
42-74181

8106300048

CF

reactor problem and actions being taken to determine the appropriate physical protection requirements for these facilities. The interim status report was published on December 19, 1979, as SECY 79-187C.

The four issues identified in SECY 79-187C and addressed in this paper are the determination of:

1. What radiation dose rate levels are needed for exemption purposes, (review the 100 rem/hr at 3 feet standard),

2. What safeguards credit should be given for fuel type and reactor design,

3. What constitutes "contiguous site" based on reasonable application of 10 CFR 73.60,

4. What safeguards credit should be given for intermediate enrichments of fuel.

Nonpower Reactor Status Report

In SECY 79-187B, 22 nonpower reactor licensees were listed as having licenses to possess a formula quantity or more of SSNM. Of those 22, seven have taken or are taking action to reduce their holdings to less than a formula quantity of SSNM and the NRC will take action to amend their licenses to reduce possession authorization below a formula quantity. These seven licensees are:

- Babcock and Wilcox, Lynchburg, Virginia
- Pennsylvania State University
- University of Missouri (Rolla)
- University of Washington
- Rensselear Polytechnical Institute
- Westinghouse, Zion, Illinois
- University of California (Los Angeles)

The remaining fifteen nonpower reactor licensees will continue to possess 5 kgs or more of highly enriched uranium (HEU) onsite and the determination of the appropriate safeguards category for each of these reactors is contingent upon the resolution of the issues addressed in this paper. These fifteen nonpower reactors are:

- General Electric, Vallecitos, California
- Georgia Tech
- Massachusetts Institute of Technology
- Union Carbide, Tuxedo, New York
- Rhode Island AEC
- University of Michigan
- University of Virginia

Exhibit K

COLLEGE
OF
ENGINEERING

DEPARTMENT OF NUCLEAR ENGINEERING SCIENCES

REPLY TO
UNIVERSITY OF FLORIDA
DEPARTMENT OF NUCLEAR ENGINEERING SCIENCES
202 NUCLEAR SCIENCES CENTER
GAINESVILLE, FLORIDA 32611
AREA CODE 904 PHONE 392-1401

50-83

October 10, 1978

Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors
Nuclear Regulatory Commission
Washington, D.C. 20555

RE: Renewal of Facility Operating License No. R-56

Dear Sir:

We are in the process of compiling all necessary and required information associated with the license renewal of the UFTR. A significant part of a new Safety Analysis Report and Technical Specifications has been done.

Presently, a parallel effort is being made, with Department of Energy support (Contract No. EY-76-S-05-4014), to change the UFTR fuel to 4.8% enriched in U235, UO_2 pellets in stainless steel cladding. This change will directly affect portions of the SAR. We respectfully request that only one license renewal with the new fuel be submitted and considered for re-licensing, rather than two consecutive and different applications.

Portions of the required additional information not affected by the change of fuel will be submitted earlier for review.

We expect that our studies on the new core performance and the safety evaluation be finished by the end of March, 1979 with final submission to the NRC by June, 1979.

The UFTR is an operating reactor, with almost 20 years of safe operating record. The studies and changes will further improve the operational capabilities and safety of the reactor.

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PDR

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Please let us know if further information is required.

Sincerely yours,

N.J. Diaz
Associate Professor &
Director of Nuclear Facilities

NJD/jcb

APPROVED:

M.J. Ohanian, Chairman

cc: L. Akers (DOE)
G.R. Dalton (UFTR Subcommittee)
C.E. Roessler (Rad. Control Committee, UF)
Steve Ramos (NRC)

Exhibit L

APPLICATION FOR A CLASS 104 LICENSE

FOR A RESEARCH REACTOR FACILITY

Based on

Code of Federal Regulations, Title 10, Part 50

to

U.S. Nuclear Regulatory Commission

R. R. O'Neill, Dean
School of Engineering and Applied Science
University of California
Los Angeles

February 1980

AMENDED: April 1982

(iii) Foreign Relationships: The applicant is in no way owned, controlled, or dominated by an alien, a foreign corporation, or foreign government.

(4) Agent: The applicant is not acting as the agent or representative of another in filing this application. The applicant is the principal party.

e. Class of license applied for:

Class 104 License.

Use to which the facility will be put:

The reactor and its supporting laboratories will be used for the education of senior undergraduate and graduate students in nuclear engineering and related sciences. In addition to formal courses and demonstrations, the reactor will be used to support research at the M.S. and Ph.D. levels.

Period of time for which license is requested:

Twenty (20) years, or until March 30, 2000.

Other licenses applied for in connection with this facility:

Special Nuclear Material: (1) 4700 gms U-235 (irradiated),
(2) 4700 gms U-235 (fresh),
(3) Pu-239 as a 2 Curie, Pu-Be neutron source.

f. Financial qualifications of the applicant:

This item is treated in Appendix I "Financial Qualifications".

g. Deleted

h. Not applicable

i. Not applicable

j. No restricted data or defense information is contained in this application or in any material offered in support of this application.

TABLE III/6-2 TRAINING REACTOR CHARACTERISTICS

	1960	1980
DATE		
TYPE	HETEROGENEOUS, THERMAL	•
POWER	10 KW	100 KW
FLUX LEVEL (AT 10 KW)	1×10^{11} N/CM^2 SEC	1.5×10^{12} N/CM^2 SEC
EXCESS REACTIVITY (TECH SPEC LIMIT)	0.58% ρ AT 32°F	2.3% ρ AT ROOM TEMP
EXCESS REACTIVITY INSTALLED	1.5% ρ AT ROOM TEMP	1.8% ρ AT ROOM TEMP
CLEAN COLD CRITICAL MASS	3200 GM U-235	•
EFFECTIVE PROMPT NEUTRON LIFETIME	1.4×10^{-4} SEC	2×10^{-4} SEC
UNIFORM WATER VOID COEFFICIENT	-0.19% ρ/% VOID	-0.164% ρ/% VOID
TEMPERATURE COEFFICIENT	-0.48×10^{-2}% ρ/°F	$-.481 \times 10^{-2}$% ρ
U-235 MASS COEFFICIENT	+0.31% ρ/% U-235 MASS	+.3% ρ/% U-235
START-UP SOURCE	2 CURIE Pu Be	6.6 MCi Ra Be
REFLECTORS	GRAPHITE (1.67 GM/CC)	•
MODERATOR	H_2O AND GRAPHITE	•
DELAYED NEUTRON FRACTION	0.0068	.0065
FUEL PLATES		
FUEL	93% ENRICHED, U-Al ALLOY	•
FUEL LOADING	3,446.2 GM U-235	3,556 GM U-235
PLATE THICKNESS	0.070 IN.	•
WATER CHANNEL	0.137 IN.	•
ALUMINUM TO WATER RATIO (VOL)	0.51	•
MEAT COMPOSITION	13.4 WT% U-Al	•
COOLANT	H_2O	•
FLOW	10 GPM	15 GPM
TEMPERATURE, IN	103°F	100°F
TEMPERATURE, OUT	110°F	140°F
CONTROL BLADES	Cd, SWINGING VANE, GRAVITY FALL	•
NUMBER	3 SAFETY: 1 REGULATING	•
INSERTION TIME	0.324 SEC (CALCULATED)	0.5 SEC (MEASURED)
REMOVAL TIME	90 SEC (MINIMUM)	100 SEC
BLADE WORTH, SAFETY	3 RODS 1.5% ρ = 4.5% ρ	3 RODS ≈ 1.6% ρ = 4.8%
BLADE WORTH, REGULATING	1 ROD 0.6% ρ = 0.6% ρ	1 ROD ≈ 1% ρ
	TOTAL = 5.1% ρ	TOTAL ≈ 5.8% ρ
REACTIVITY ADDITION RATE, MAX.	0.02% ρ/SEC	.05% ρ/SEC
SHIELD (CONCRETE)		
SIDES, CENTER	6 FT. 0 IN. CAST, MAGNETITE	•
SIDES, SHIELD TANK END	6 FT. 8 IN. CAST, ORDINARY	•
SIDES, THERMAL COLUMN END	6 FT. 8 IN. CAST, MAGNETITE	•
MIDDLE	CAST CONCRETE BLOCKS	•
ABOVE CORE	5 FT. 10 IN. MAGNETITE BLOCKS	• PLUS 39" OF BORATED PARRAFIN
ENDS	3 FT. 4 IN. MAGNETITE BLOCKS	•
EXPERIMENTAL FACILITIES		
THERMAL COLUMN, HORIZONTAL	5 FT. x 5 FT. x 4 FT. 11 IN. LONG	60 IN. x 52 IN. x 43 IN. LONG REMOVABLE
THERMAL COLUMN, VERTICAL	PROVISION FOR INSTALLATION	•
SHIELD TEST TANK	5 FT. x 5 FT. x 14 FT. 6 IN. HIGH	•
EXPERIMENTAL HOLES	2 - HORIZONTAL, 6 IN. DIAMETER	•
	5 - HORIZONTAL, 4 IN. DIAMETER	4 - HORIZONTAL, 4 IN. DIAMETER
	3 VERTICAL, 1½ IN. DIAMETER	3 - VERTICAL, 1-7/8 IN. DIAMETER
EXPERIMENTAL HOLES, THERM. COL.	15 REMOVABLE GRAPHITE STRINGERS	•
FOIL SLOTS	11 - HORIZONTAL, 1/8 IN. x 1/2 IN.	•
	16 - VERTICAL, 3/8 IN. x 1 IN.	•

4/30/82

Exhibit M

ANL-6285
Reactor Technology
(TID-4500, 16th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

ARGONAUT REACTOR DATABOOK

A compilation of experimental and theoretical results of
work done with, or related to, the Argonaut Reactor
to July 1960

by

W. J. Sturm and D. A. Daavettila

January 1961

Operated by The University of Chicago
under
Contract W-31-109-eng-38

Section A

CHARACTERISTICS OF THE ARGONAUT

The Argonaut Reactor was designed for training in both nuclear engineering and research, and the experience of nearly four years of operation has proved the design to be practical. The reactor, because it is simple to operate and extremely safe, is well suited for training people without previous reactor experience. Safety is a primary design feature. As a research tool, the usefulness of the reactor is enhanced by the fact that the core is readily accessible and that the core geometry is flexible. A graphite thermal column and a large water tank are integral parts of the reactor, and numerous types of experiments can be done in these media.

The 10-kw maximum operating power of the reactor prohibits certain types of experiments, but this disadvantage is far outweighed by the fact that fuel does not become a serious radiation hazard. For all the experiments whose results are presented in this compilation, the operating power was less than 100 watts and for most less than 10 watts.

This section lists some general nuclear and engineering data of the Argonaut Reactor in order to present the basic design. The data cover only the main points of a broad area, but this will be expanded in later sections. The nuclear data given in this section are the result of the first theoretical calculations and preliminary critical studies.

A.1. General (Ref. I-7)

Type:	Training reactor
Design power:	10 kw
Normal operating power:	~100 watts
Normal operating schedule:	8 hours a day, 5 days a week
Principal uses of reactor:	Education and training

A.2. Fuel

Nominal fresh loading:	1 slab: 2.0 kg U^{235} 2 slabs: 3.6 kg U^{235} 3-in. annular: 4.0 kg U^{235}
Total fuel inventory:	6 kg U^{235}
Fuel element shape:	24 x 2.84 x 0.098-in. plates
Fuel mixture:	39 w/o Al, 7.8 w/o $U_3^{235}O_8$, 31.2 w/o $U_3^{238}O_8$; Al matrix.
Fuel dimensions:	24 x 2.84 x 0.094 in.
Cladding thickness:	0.002 in. (avg)
Cladding material:	Aluminum
Type of subassembly:	Stacked parallel plates
No. of elements per subassembly:	17
Subassembly dimensions:	6 x 3 x 24 in.
Normal number of subassemblies in core:	1 slab: 6-9 subassemblies 2 slabs: 12 subassemblies 3-in. annular: 24 subassemblies
Normal arrangements of subassemblies:	1 slab, 2 slabs, or full circle in cylindrical annulus.
Normal lifetime of standard subassemblies:	Indefinite

A.3. Reactor

Overall active core dimensions:	1 slab: Annular sector - 30 in. OD, 24 in. ID, 24 in. high, subtending a 90° angle. 2 slabs: 2 of above, diametrically opposed.

Exhibit N

UNIVERSITY OF CALIFORNIA, LOS ANGELES UCLA

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OFFICE OF THE CHANCELLOR
LOS ANGELES, CALIFORNIA 90024

August 26, 1982

Mr. John H. Bay, Esq.
Embarcadero Center
Twenty-Third Floor
San Francisco, California 94111

Dear Mr. Bay:

In response to our agreement reached over the telephone on August 18, 1982 and recorded in your letter to me of the same date, I have enclosed the following information:

- a table representing the fuel inventory by various category at the UCLA facility since 1970 contained in memo, Ostrander to Cormier; and

- answers to the written questions on the "Fuel Self-Protection Calculations" which you had hand-delivered to my office on August 23rd; these questions were essentially follow-up questions to our interrogatory responses of August 9th.

I trust that you will find our responses to your discovery requests both complete and timely.

Very truly yours,

William H Cormier

William H. Cormier
UCLA Representative

Enclosure

cc: Service List

MEMORANDUM

25 August 1982

TO: W. Cormier
2241 Murphy

FROM: N. Ostrander
2567 Boelter Hall

SUBJ: NEL Fuel Inventory Since 1970

I have constructed the attached inventory record for your response to Mr. Bay's request of August 18, 1982. Inventorial practices have changed over the several AEC-ERDA-NRC administrations and even within the lifetime of the NRC. The general trend has been to add detail by distributing inventory into an increasing number of categories. Descriptive words have been replaced by a three symbol code. There have been several generations of such codes, and no assurance that they are one-for-one translatable. For example, one can translate "encapsulated, enriched, unirradiated, uranium-alloy scrap" into the category "uranium" but the inverse transformation is not possible.

All of this goes to say that I have made a best effort to provide a complete record, but I have had to make some interpretations based upon continuity of category by continuity of numbers. I cannot attest to the absolute accuracy of the record. I think it is a reasonable, but not necessarily unique interpretation of the available records.

INVENTORY OF U-235 ISOTOPE IN FUEL, kg

DATE	Irradiated Fuel		Fresh Fuel		TOTAL
	In-Core	In Pits	Useful	Scrap	
3-31-70	3.50	-	-	0.02	3.52
6-30-71	3.50	-	2.53	0.02	6.05
12-31-71	3.56	0.73	3.74	0.94	8.97
12-21-74	3.55	0.73	3.74	0.60	8.62
9-30-80	3.53	-	3.74	0.60	7.87
9-30-81	3.53	-	3.75	-	7.28
8-25-82	3.53	-	1.39	-	4.92

Except for the small burn-up (~ 1 gm per year), the inventories are constant over any interval between adjacent dates. E.g., from 12-31-74 to 9-30-80, the total inventory was approximately 8.62 kilograms. The dates are inventorial record dates and not the actual dates of the material transfer.

~~Exhibit "B"~~
Exhibit "C"

October 28, 1974

Karl R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing
USAEC
Washington, D. C. 90545

Dear Sir:

Due to the sensitive nature of the contents of this letter, we request that this document be withheld from public disclosure pursuant to Section 2.790 of 10 CFR Part 2.

Upon redoing our calculations on the Special Nuclear Material inventory, we found that our scrap quoted to you was the total uranium content, not the U-235 content. Therefore, we have at our facility a total SNM inventory of 9.387 kg. Of this, 4.293 kg. are exempt and 5.094 kg. are non-exempt.

In order to comply with the 5 kg. limit and approval of our security system, we request permission to ship 340 grams of U-235 to Oak Ridge - Y-12 facility. This would bring our non-exempt SNM inventory down to 4.754 kg. and our total SNM inventory down to 9.047 kg.

Forms OR-658C and Forms OR-658A have been sent to:
Joe Mahler
Product Division
USAEC
Oak Ridge Operations Office
P. O. Box "E"
Oak Ridge, Tennessee 37831

Sincerely,

Charles [illegible]ugh III
Reactor Sup[illegible]sor

UNIVERSITY OF FLORIDA TRAINING REACTOR

HAZARDS SUMMARY REPORT

Prepared by

J. M. Duncan

$5.00 per copy

A Report to
The United States Atomic Energy Commission
Division of Civilian Application

From the
Department of Nuclear Engineering
College of Engineering
University of Florida
Gainesville, Florida
October, 1958

The biological shield is made of cast-in-place concrete with sections of barytes concrete carefully located to reduce the overall shield thickness. Access to the ends and top of the reactor is provided by removal of ordinary concrete blocks cast to fit the openings.

These blocks, weighing up to 4500 lb., each, have pick-up plugs so that they may be handled by means of the overhead crane. The concrete is thick enough to reduce the radiation leaking from the reactor to 0.7 mr/hr at the full operating power of 10 kw.

The nuclear characteristics of this reactor, given in Table 4.5A, are similar to those of other water-moderated reactors using similar fuel plates such as the LITR, MTR, BSTF, Borax I, II, and III, and Argonaut.*

4.5.1 Reactor Core

The reactor core consists of 24 bundles of fuel plates and 12 single fuel plates contained in six water-filled aluminum boxes surrounded by reactor-grade graphite. Four cadmium control blades, protected by magnesium shrouds, move between the fuel boxes.

The fuel plates are in the form of the MTR type (Figure 4.5F). A sheet of 0.040-in.-thick 20 per cent enriched uranium-aluminum alloy is completely clad with 0.015-in. thickness of aluminum. These plates are 25 5/8 in. long, 2 7/8 in. wide, and have a total thickness of 0.070 in. Each plate contains approximately 14.5 grams of uranium-235. These plates are bolted into bundles of eleven plates each, spaced on 0.207-in. centers, leaving 0.137-in. channels between plates. In each fuel box there is space for four fuel bundles and two single plates. When fully loaded in this manner the six fuel boxes contain 276 plates with a total of approximately four kilograms of uranium-235.

The calculated cold clean critical mass of the reactor is 3.5 kilograms of U-235. In order to adjust the fuel loading to achieve the specific excess k desired for operation of the reactor, aluminum dummies may be substituted for fuel plates in assembling the fuel bundles. An estimate of the worth of a single plate lies between 0.1 and 0.2 per cent k, which should allow sufficient flexibility so that no special or partial plates will be required initially for adjusting reactivity.

Since heat-transfer considerations are of minor importance for this reactor, a number of different fuel elements could be considered. It is desirable, however, to use a structure which closely resembles those used in the Borax reactors, since the behavior of Borax reactors during power excursions has been experimentally demonstrated. The use of metallic fuel plates of high thermal conductivity minimizes the extrapolation of these data so that there is a greater degree of confidence in the calculations of the results in the unlikely event of an excursion.

Plates of 20 per cent enriched uranium-aluminum alloy jacketed in aluminum have been selected for the initial loading because (1) less stringent security requirements are associated with this enrichment, (2) nuclear characteristics are satisfactory for the purpose, and (3) proven fuel plates of this enrichment are available.

The six type-1100 aluminum fuel plate boxes have inside dimensions of 5 in. by 6 in. by 48 in. high (Figure 4.5G). The plates rest on a supporting member, 1½ in. above the bottom of the box, which centers the fuel vertically in the reactor and provides for a water reflector above and below the plates. The aluminum boxes are connected at the bottom by means of an aluminum header through which the cooling water is supplied. The tops of the boxes are connected by aluminum overflow and vent pipes. Each box rests in a rectangular hole in the graphite prism and, if desired, can be removed by first unloading the fuel plates — then, disengaging four nuts from studs on the bottom flange with a long-handled wrench.

The top of each box is closed by a plug which extends upward through the graphite which forms the base for the vertical thermal column. The upper part of the plug

* "Summary Report on the Hazards of the Argonaut Reactor," D. H. Lennox and C. N. Kelber, ANL-5647.

emphasis added

Exhibit Q

ANL-5647
SPECIAL

ARGONNE NATIONAL LABORATORY
P. O. Box 299
Lemont, Illinois

SUMMARY REPORT ON THE HAZARDS OF THE ARGONAUT REACTOR

by

D. H. Lennox and C. N. Kelber

Including work done by: R. H. Armstrong
W. L. Kolb
Andrew Selep
B. I. Spinrad

Reactor Engineering Division

December, 1956

Operated by The University of Chicago
under
Contract W-31-109-eng-38

RE-6-17775-A
W.KOLB:P.D.,12-3-56

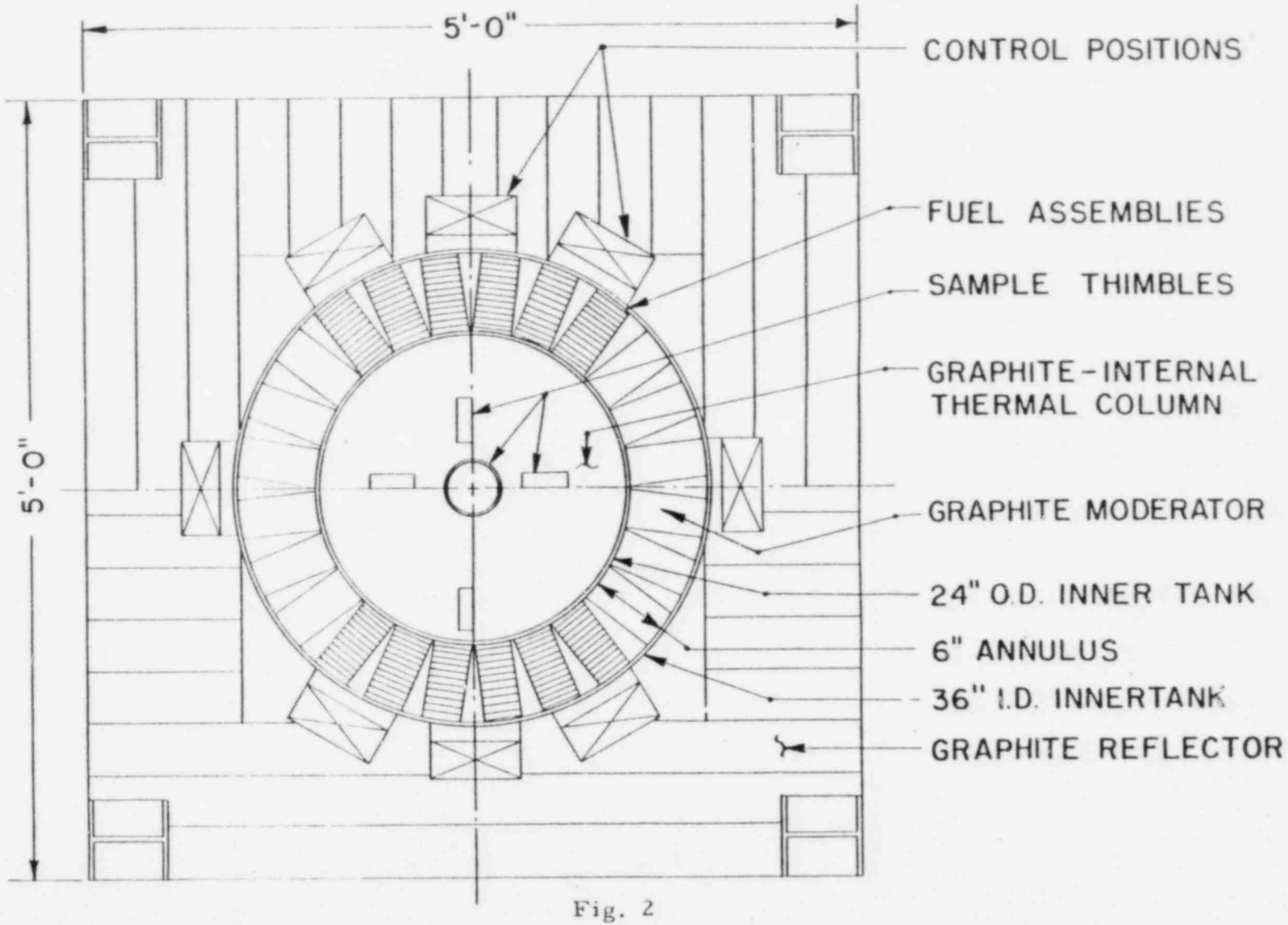


Fig. 2

PLAN SECTION OF CORE LATTICE AND REFLECTOR

5. Graphite Core Pieces

The core contains 24 graphite wedges, each $1\frac{5}{8}$ in. at the base, 6 in. thick, 24 in. high, tapering to a blunt point. It also contains twelve graphite dummy blocks (3 x 6 x 24 in.).

When water is admitted to the fuel region the graphite filler pieces become submerged and hence must be waterproofed. As an inexpensive substitute for aluminum cladding, an aluminum-Krylon plastic spray-coat is used. Irradiation in CP-5 comparable to several years of operation of Argonaut caused no degradation of the coating.

6. Fuel Elements

Each complete fuel assembly box contains 17 aluminum-clad plates (Fig. 7). The over-all dimensions are 6 in. x 3 in. x 24 in. long. The plates are assembled with aluminum bolts at top and bottom. Dummy aluminum plates or graphite slabs can be substituted for fuel plates to vary the quantity of fuel per box. Spacing between plates is maintained by two Teflon washers (1/4 in. thick) attached to each end of the individual plates. This separation gives a metal to H_2O volume ratio of 0.4.

An inexpensive fabrication technique for making fuel plates containing 35 wt-% of 20% enriched U_3O_8 was developed by the Argonne Metallurgy Division. A hot extrusion of a mixture of U_3O_8 and 2S aluminum powder gives plates with negligible void volume and over-all dimensions of 0.098 in. thick by 24 in. long and 2.84 in. wide.

Aluminum powder and U_3O_8 in the proper ratio were placed in a $3\frac{1}{2}$-in. diameter vented aluminum can, heated to 483C, sealed and then extruded in a 400-ton horizontal press. The resulting fuel sheet, approximately 17 ft long, was cut into sections 2 ft long. A clad averaging 2 mils thick covered the plate except on the ends at the point of cutoff and at some scratch points along the surface. Exposed portions of the fuel matrix present no corrosion problems; however, a plastic spray is applied to stop fission recoils.

The uranium oxide content of each plate varies; those cut from the ends of the extrusion contain somewhat less U_3O_8 than the average. The composition of each plate is:

U^{235}	19.6 gm $\pm$ 10%
U_3O_8	114 gm
Al	248 gm

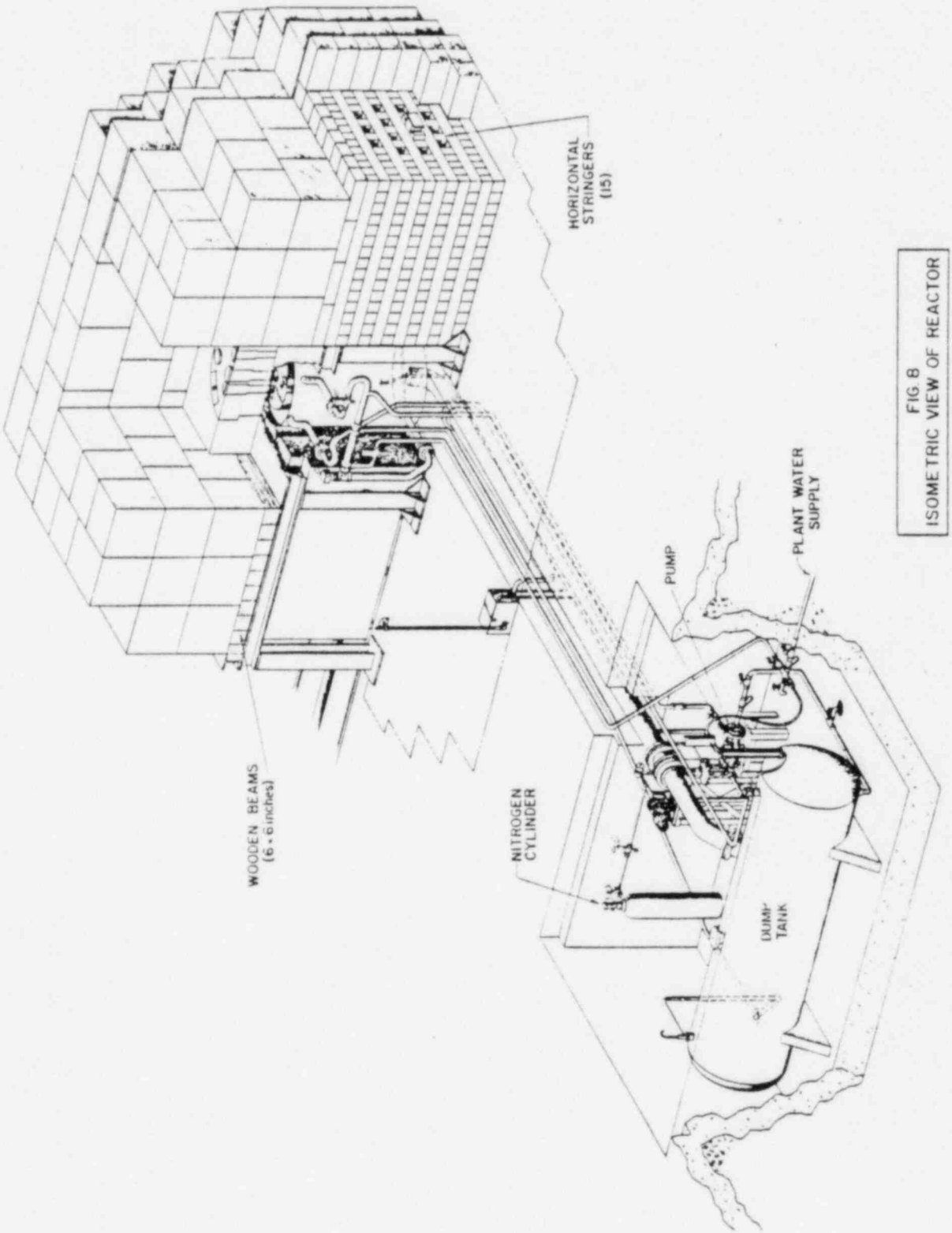


FIG 8
ISOMETRIC VIEW OF REACTOR

11. Start-up Source

An antimony-beryllium-photoneutron source is used to provide neutrons for start-up and multiplication measurements. The source is motor driven from a loading port outside the concrete shield in a trench under the reactor tank.

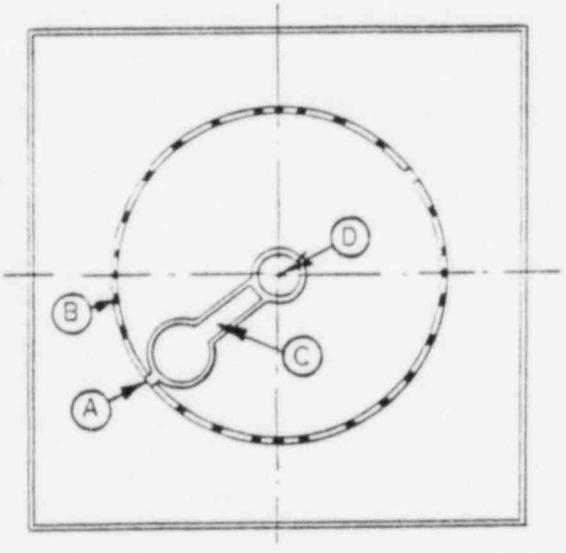
The antimony in the source is removable from the beryllium to permit rejuvenation in CP-5. An activity of ~5 x 10^8 neutrons/second is obtained from a solid cylinder of antimony 1 in. OD x 1½ in. long. The antimony is clad with aluminum, irradiated for 5 days in CP-5, and inserted in a 3-inch cube of beryllium.

12. Handling Equipment

A jib-type crane is installed in the floor within the reactor shield so that the jib arc reaches all blocks. The rated capacity is 1/2 ton at the end of the boom and 4700 pounds at a point 6 ft from the mast, corresponding to a position directly over the top shield plug. A portable lead coffin is used for transferring either fuel elements or antimony from the start-up source.

a. Top Shield Plug

A steel-clad, barytes concrete-filled slab (61⅜ in. x 61⅜ in. x 1 ft thick) shields the top of the active region (see illustration).



LEGEND

A Index Key

B Index Slot

C Access to Fuel and Four Radial Experimental Holes

D Central Experimental Port

D. Experimental Facilities

Space and structural strength is provided for exponential experiments laid on top of the core region. Removal of the upper shield plug leaves a five-foot square distributed neutron source, which may be shaped by addition of a graphite pedestal. Performance of such experiments temporarily precludes any access to the core.

A tunnel (4 x 5 ft) penetrates one side of the shield and is served with a movable cart. Initially, a water-filled tank will be mounted on the cart, plugging the tunnel. The tank may be used for (1) shielding studies; (2) water-moderated exponential measurements; or (3) solid materials may be located on the cart for migration measurements. Interlocks requiring both that the cart be completely forward and that the biological shielding be adequate before start-up can proceed ensure that cart motions cannot add reactivity to the system and that loss of water in the tank cannot lead to over-exposure of personnel.

The internal reflector has five removable vertical stringers at varying radii. Access to the stringers is through ports in the top shield plug. When these stringers are removed, samples or experimental liners must be in place before operation is permitted. Electrical interlocks ensure this condition.

Two holes (4 x 4 in.), provided by removal of concrete-graphite plugs, penetrate the shield and reflector at the active lattice midplane. The holes extend to the outer reactor tank at points 90 degrees from the external thermal column and the irradiation cart.

The external thermal column has fifteen removable stringers.

Complete removal of the internal tank is possible when the fuel annulus is unloaded. This leaves a three-foot diameter, graphite-reflected tank in which multiplication experiments may be performed; or critical experiments may be performed therein after an additional hazards review. Such review is also required for performance of internal exponential experiments, which require removal of the inner tank before replacement of the inner thermal column. The inner thermal column cannot, by its design, be unloaded while the inner tank is in the reactor.

E. Fuel Storage

The total inventory of U^{235} in the reactor building is 5.3 kg contained in fuel plates. Approximately 3.75 kg are normally contained in the reactor; the remainder are locked in a four-drawer, cadmium-lined, combination-locked file. All storage criteria have been checked to ensure against achieving criticality by flooding or other accident.

Exhibit R

ARGONNE NUCLEAR ASSEMBLY FOR UNIVERSITY TRAINING

PURPOSE: Research and Training

DATE OF INFORMATION: October 1962

GENERAL

1. Reactor type	Enriched (20%) uranium, light water moderated and cooled, graphite reflected	5. Owner and operator	Owned and operated by Argonne National Laboratory
2. Nominal reactor power	Design 10 kW thermal Normal operating power 1—100 W thermal	6. Designer and builder	Argonne National Laboratory
3. Purpose	Neutron beam source, exponential experiments, shielding studies, migration measurements, fuel studies, irradiation studies, reactor kinetics studies, general reactor properties	7. Present status & construction schedule	In operation Start of construction 1956 Reactor critical Feb. 1957
4. Location	Argonne National Laboratory, Lemont, Illinois, USA		

REACTOR PHYSICS

8. Neutron energy and lifetime	Thermal Lifetime about 1.8×10^{-4} sec	10. Neutron flux	At 10 kW: Thermal av. 1.47×10^{11} n/cm² sec Thermal max. 1.69×10^{11} n/cm² sec Fast max. 1.35×10^{10} n/cm² sec
9. Core parameters	One slab core: $\eta = 2.05$ $\varepsilon = 1.0$ $f = 0.80$ $p = 0.98$ $k_\infty = 1.60$ $k_{eff} = 1.005$ $L^2 = 3.929$ cm² $\tau = 61.3$ cm² $B^2 = 0.008$ cm^{-2} Thermal leakage factor 0.97 Fast leakage factor 0.65	11. Reactivity balance	Max. built in (cold, clean): 0.5%, used for experiments

CORE

12. Shape and dimensions	Cylindrical annulus, 36 in. od., 24 in. id., 24 in. high Various core configurations possible, consisting of 6 to 24 subassemblies. In each case the remaining annular region is filled with graphite filler blocks.	18. Average power density in core	Normal 14 W/liter (one slab core, 100 W) Design max. 1.4 kW/litre (one slab core, 10 kW)
		19. Burnup	Negligible
13. No. of channels & subassemblies	Room for 24 subassemblies in annulus One slab core — 6 subassemblies in a 90° section of annulus Two slab core — two such 90° sections diametrically opposed Annular core — every position loaded, but only the inner 3 in. with fuel	20. Fuel loading and unloading	Under normal conditions fuel is manually transferred without need of shielding. For extreme cases a jib-type crane and lead coffins with means for indexing and grappling may be used.
14. Lattice	24 positions spaced equally on a circle of a radius of 15 in.	21. Irradiated fuel storage	Cylindrical holes, 8 in. diam. and 4 ft deep in cadmium-lined concrete. Normal loading of 6 subassemblies, but it is possible to store up to 18 subassemblies.
15. Critical mass	One slab core 1.9 kg U^{235} Two slab core 3.4 kg U^{235} Annular core 4.1 kg U^{235}	22. Moderator	200 gal. light water, cooled for operation above 1 kW to room temperature
16. Core loading at rated power	One slab core 1.95 kg U^{235} Two slab core 3.5 kg U^{235} Annular core 4.2 kg U^{235}		
17. Average specific power in fuel	Normal 50 W/kg U^{235} (one slab core, 100 W) Design max. 5 kW/kg U^{235} (one slab core, 10 kW)	23. Blanket gas	None

FUEL ELEMENT

24. Form and composition	Rectangular plates, plates overall 0.098 × 2.84 × 24 in. Enrichment 20%, 35 wt. % U_3O_8 in aluminium powder matrix, hot extruded	25. Cladding	Aluminium, bonded to meat by sintering. Final cladding thickness is 0.002 in., except at end points of cutoff which are epoxy resin coated
		26. Subassemblies	17 parallel plates, spaced 0.25 in. apart, form a fuel subassembly, 6 × 3 × 24 in. overall. Dummy aluminium plates or graphite slabs may be used to vary fuel load per subassembly

SIEMENS ARGONAUT REACTOR No. 1

PURPOSE: Research and Training

DATE OF INFORMATION: August 1962

GENERAL

1. Reactor type	Argonaut type, highly enriched (19.8%) uranium, light water moderated and cooled, graphite reflected	5. Owner and operator	Siemens-Schuckertwerke AG, Erlangen
2. Nominal reactor power	1 kW thermal, continuous 10 kW thermal, max.	6. Designer and builder	Siemens-Schuckertwerke AG, Erlangen
3. Purpose	Neutron physics, lattice experiments, shielding experiments, power reactor development	7. Present status & construction schedule	In operation Start of construction: Oct. 1958 Start of assembly: Jan. 1959 Reactor critical: June 1959
4. Location	Garching (10 km North of Munich), Bavaria, Germany		

This reactor is similar to the ARGONAUT reactor at Argonne National Laboratory, USA, in its main parts with the following modifications:

Neutron flux: Thermal max. 1.4×10^{11} n/cm² sec; Fast max. 2.4×10^{11} n/cm² sec

Av. specific power in fuel: 0.5 kW/kg U^{233} at 10 kW for annular loading

Irradiated fuel storage: 18 steel clad holes in concrete floor

Coolant mass flow rate: Max. 11 liters/min

Control rods:

3 shim-safety blades	178 x 177 x 0.76 mm
1 coarse regulating blade	178 x 177 x 0.76 mm
1 fine regulating blade (A)	178 x 102 x 0.76 mm
1 fine regulating blade (B)	178 x 51 x 0.76 mm
Worth of shim safety blades	4 to 8% $\frac{\Delta k}{k}$ depending on core configuration
Worth of coarse regulating blade	3% $\frac{\Delta k}{k}$
Worth of A fine regulating blade	2% $\frac{\Delta k}{k}$
Worth of B fine regulating blade	1% $\frac{\Delta k}{k}$
Speed of blades	0.25 cm/sec = 15×10^{-3} % $\frac{\Delta k}{k}$/sec (for coarse regulating blade)

Containment: No actual containment. The reactor is located in a semi-airtight concrete hall, 16 x 12 m, 8 m high

Surroundings: Sparcely populated farmland on the Isar river, about 10 km north of Munich

Bibliography:

1. Atomwirtschaft, 11, 479 (Nov. 1959)
2. Siemens-Zeitschrift, No. 12, p. 745 (1959)

RESEARCH FACILITIES

Designation	No.	Position	Useful dimensions (cm)	Neutron flux (n/cm²sec)	Remarks
Horizontal neutron beams	2	(1)	7.9 x 7.9		
Horizontal thermal column	1	(2)	126 x 156 157 long		
Removable stringers of horizontal thermal column	15	(3)	10.5 x 10.5		
Vertical thermal column	1	(4)	61 diam.		
Removable stringers of vertical thermal column	1 4	(5) (6)	10.3 diam. 8.2 x 3.1		
Mobile water tank	1	(7)	126 x 107 218 long		

SIEMENS ARGONAUT REACTOR KARLSRUHE

PURPOSE: Research and Training

DATE OF INFORMATION: August 1962

GENERAL

1. Reactor type	Argonaut type, highly enriched (19.8%) uranium, light water moderated and cooled, graphite reflected	5. Owner and operator	Gesellschaft für Kernforschung mbH.
2. Nominal reactor power	10 W thermal, continuous	6. Designer and builder	Consortium of Siemens-Schuckertwerke A. G., Lurgi G. m. b. H., Pintsch-Bamag A. G.
3. Purpose	Neutron physics, research in radio-chemistry and educational purposes	7. Present status & construction schedule	In operation Start of construction — Oct. 1961 End of construction — Oct. 1962 Reactor critical — Jan. 1963
4. Location	Leopoldshafen, near Karlsruhe, Germany		

This reactor is similar to the ARGONAUT reactor at Argonne National Laboratory, USA in its main parts with the following modifications:

Neutron flux: Thermal max. 1.4×10^{8} n/cm² sec
Fast max. 2.4×10^{8} n/cm² sec

Av. specific power in fuel: 0.5 W/kg U^{235} for annular loading

Irradiated fuel storage: 30 steel clad storage holes in concrete slab

Coolant mass flow rate: Max. 2 kg/sec

Control rods:

3 shim-safety blades	175 × 177 × 0.76 mm
1 coarse regulating blade	178 × 177 × 0.76 mm
1 fine regulating blade (A)	178 × 102 × 0.76 mm
1 fine regulating blade (B)	178 × 51 × 0.76 mm

Worth of coarse regulating blade: $3\% \frac{\Delta k}{k}$

Worth of A fine regulating blade: $2\% \frac{\Delta k}{k}$

Worth of B fine regulating blade: $1\% \frac{\Delta k}{k}$

Speed of blades: 0.25 cm/sec = 15×10^{-3} % $\frac{\Delta k}{k}$/sec (for coarse regulating blade)

Containment: Gas-tight spherical hall with flat dome, max. diam. 17 m, height 10 m

Surroundings: Forests, 13 km north of Karlsruhe in Rhine Valley

radius from reactor centre	population
2.4 km	0
4 km	10 000
8 km	45 000
20 km	530 000

Bibliography:

1. Atomwirtschaft, 11, 479 (Nov. 1959)
2. Siemens-Zeitschrift, No. 12, p. 745 (1959)

RESEARCH FACILITIES

Designation	No.	Position	Useful dimensions (cm)	Neutron flux (n/cm² sec)	Remarks
Horizontal neutron beams	2	(1)	7.9 × 7.9		
Horizontal thermal column	1	(2)	126 × 156 157 long		
Removable stringers of horizontal thermal column	15	(3)	10.5 × 10.5		
Vertical thermal column	1	4	61 diam.		
Removable stringers of vertical thermal column	1 4	(5) (6)	10.5 diam. 5.2 - 3.1		
Mobile water tank	1	(7)	126 - 107 218 long		

UNIVERSITY OF FLORIDA TRAINING REACTOR

PURPOSE: Research and Training

DATE OF INFORMATION: April 1963

GENERAL

1. Reactor type	Argonaut type, enriched (20% and 93%) uranium, light water moderated and cooled, graphite reflected. Unless otherwise stated, data listed is for 20% enr. fuel (see Remarks)	5. Owner and operator	University of Florida
2. Nominal reactor power	100 kW thermal	6. Designer and builder	Designed by General Nuclear Engineering Corp. Built by AMF Atomics (American Machine and Foundry Co.)
3. Purpose	Isotope production and training of graduate students in reactor physics, dynamics, shielding	7. Present status & construction schedule	In operation Start of construction Dec. 1957 Reactor critical May 1959
4. Location	Gainesville, Florida, USA		

REACTOR PHYSICS

8. Neutron energy and lifetime	Thermal Lifetime 1.4×10^{-4} sec. (design), $\sim 3 \times 10^{-4}$ sec. (measured)	10. Neutron flux	Thermal max. 1.6×10^{12} n/cm² sec Epi-cadmium max. 2.1×10^{12} n/cm² sec
9. Core parameters	$\eta = 2.08$ $\epsilon = 1.0$ $f = 0.790$ $p = 0.940$ $k_{\infty} = 1.545$ Fast leakage factor 0.681 Thermal leakage factor 0.950	11. Reactivity balance	Max. built-in (cold, clean) 0.6% To compensate for temperature 0.20% experiments 0.15% Xenon 0.20% burnup 0.20%

CORE

12. Shape and dimensions	Rectangular prism; overall approx. 22 × 20 in., 24 in. high	18. Average power density in core	1.4 kW/litre
		19. Burnup	0.14% of fissionable material
13. No. of channels & subassemblies	Two banks of 3 fuel boxes each, separated by 12 in. graphite. Fuel boxes are each 5 × 6 × 48 in. high (inside dimensions) with fuel support plate 11.5 in. from bottom of box. 4 fuel subassemblies plus 2 single fuel plates in each box.	20. Fuel loading and unloading	Fuel loading and unloading systems consists of a lead-steel-concrete transfer cask, positioning plates, overhead crane, mirrors and various handling tools.
14. Lattice	Distance between fuel boxes in a bank is 1 in. Distance between banks is 12 in. Center to center of fuel plates 0.207 in.	21. Irradiated fuel storage	20% enr. fuel is kept in 27 storage pipes in concrete floor, each 4 ft deep, 4 in. diam. 93% enr. fuel is stored in a normal steel safe since it is only slightly radioactive (see Remarks).
15. Critical mass	3.6 kg U^{235} in case of 20% enriched fuel 3.1 kg U^{235} in case of 93% enriched fuel	22. Moderator	Light water in the fuel boxes. Graphite (density 1.6) between the fuel boxes.
16. Core loading at rated power	3.6 kg U^{235} for 20% enriched fuel 3.2 kg U^{235} for 93% enriched fuel		
17. Average specific power in fuel	Approx. 28 kW/kg U^{235}	23. Blanket gas	None

FUEL ELEMENT

24. Form and composition	Rectangular plates, picture frame technique. Plate overall 0.07 × 2.875 × 25.625 in. Enrichment 20% or 93% U^{235} in Al, ave. 93% enr. fuel elements have the same overall dimensions and plates per bundle as 20% enr. fuel elements.	25. Cladding	0.015 in. aluminum
		26. Subassemblies	17 curved plates with a 0.157 in. water gap between each; one complete subassembly 4 subassemblies per fuel box

AEG PRÜFREAKTOR PR-10

PURPOSE: Research

DATE OF INFORMATION: June 1962

GENERAL

1. Reactor type	Argonaut type, enriched (20%) uranium, light water moderated and cooled, graphite reflected	5. Owner and operator	Allgemeine Elektricitäts-Gesellschaft, Frankfurt am Main
2. Nominal reactor power	10 W thermal	6. Designer and builder	Allgemeine Elektricitäts-Gesellschaft, Frankfurt am Main
3. Purpose	Reactor physics for power reactor design	7. Present status & construction schedule	In operation Start of construction Oct. 1959 Reactor critical Jan. 1961 Full power operation Feb. 1961
4. Location	Near Gross-Welzheim am Main, Unterfranken, Bavaria, Germany		

REACTOR PHYSICS

8. Neutron energy and lifetime	Approx. 0.026 eV Lifetime $1.9 \cdot 10^{-4}$ sec	10. Neutron flux	Thermal av. radial 1.75×10^8 n/cm² sec av. axial 2.0×10^8 n/cm² sec max. 2.5×10^8 n/cm² sec Fast av. radial 2.75×10^8 n/cm² sec av. axial 1.9×10^8 n/cm² sec max. 4.15×10^8 n/cm² sec
9. Core parameters	$\eta = 2.021$ $\varepsilon = 1.01$ $f = 0.812$ $p = 0.97$ $k_\infty = 1.61$ $\tau = 90$ cm² $L^2 = 6.7$ cm²	11. Reactivity balance	Max. built in: 0.6% for experiments

CORE

12. Shape and dimensions	Annular core in shape of an octagon; alternatively 80 cm (31.5 in.) high, 61.8 cm (24.2 in.) id., 79.2 cm (31.2 in.) od. or 80 cm (31.5 in.) high, 91.8 cm (36.2 in.) id., 108.8 cm (42.8 in.) od.	18. Average power density in core	Not available
		19. Burnup	Not available
13. No. of channels & subassemblies	Outer annulus consists of 24, inner annulus of 16 aluminium boxes located in vertical holes in graphite. One fuel sub-assembly per box	20. Fuel loading and unloading	Manual transfer using refuelling coffin and overhead crane
14. Lattice	Not available	21. Irradiated fuel storage	36 vertical tubes in floor with concrete plugs
15. Critical mass	6.4 kg U^{235} for 24 subassembly configuration	22. Moderator	Approx. 100 liters light water, max. temp. 35° C Graphite between fuel boxes
16. Core loading at rated power	6.4 kg U^{235}		
17. Average specific power in fuel	1.56 W/kg U^{235}	23. Blanket gas	None

FUEL ELEMENT

24. Form and composition	Rectangular plates produced by picture frame technique Meat dimensions 2.4 × 75 × 750 mm (0.095 × 2.95 × 29.5 in.) Plate overall 3 × 80 × 800 mm (0.118 × 3.15 × 31.5 in.) Enrichment 20% U_3O_8 in aluminium matrix	25. Cladding	Aluminium 0.3 mm (0.012 in.) thick
		26. Subassemblies	[illegible] plates form a fuel subassembly; ratio of active to dummy plates variable

CORE HEAT TRANSFER

27. Heat transfer area	33.8 m² (for 24 subassembly configuration)	32. Coolant mass flow rate	Not applicable, nat. convection
28. Heat flux	Negligible	33. Coolant pressures & temperatures	Slightly above ambient
29. Fuel element temperatures	Not available	34. Hot channel factors	Not available
30. Heat transfer coefficient	Not available		
31. Coolant flow area & velocity	Approx. 900 cm² (for 24 subassembly configuration)	35. Shut-down heat removal	No provision

CONTROL

36. Control, regulating and safety rods	2 shim-safety blades, active length 20 cm (8 in.) 2 regulating blades, active length 10 cm (4 in.): "A" blade active width 3 cm (1.15 in.) "B" blade active width 1.5 cm (0.59 in.) 4 shut-down blades All blades are Cd. clad with aluminium, 150 cm (63 in.) long Worth of shim-safety blades 1.2 % $\frac{\Delta k}{k}$ Worth of "A" blade 0.4 % $\frac{\Delta k}{k}$ Worth of "B" blade 0.13 % $\frac{\Delta k}{k}$ Worth of shut-down blades 2.4 % $\frac{\Delta k}{k}$ Speed of shim-safety blades 0.5 cm/sec ≈ 6×10^{-1} % $\frac{\Delta k}{k}$/sec Speed of "A" regulating blade 0.5 cm/sec ≈ 4×10^{-1} % $\frac{\Delta k}{k}$ sec Speed of "B" regulating blade 0.5 cm/sec ≈ 1.3×10^{-1} % $\frac{\Delta k}{k}$ sec	38. Scram time & mechanism	Magnetic clutch, gravity fall Delay time 0.06 sec Rod travel time 0.3 sec
		39. Sensitivity of auto. control	No automatic control
		40. Temperature coefficients	-6×10^{-1} % $\frac{\Delta k}{k}$/°C
		41. Burnable poison	None
		42. Other control, safety & shut-down provisions	Boric acid injection foreseen
37. Reactivity addition rate	Max. 6×10^{-1} % $\frac{\Delta k}{k}$/sec		

REACTOR VESSEL & OVERALL DIMENSIONS

43. Form, material and dimensions	Aluminium fuel boxes, each containing one subassembly (See No. 13)	44. Working, design & test pressures	Atmospheric
		45. Reactor with shielding	4.5 × 4.5 m, 2.5 m high (14.8 × 14.8 ft, 8.2 ft high)

REFLECTOR AND SHIELDING

46. Reflector	Graphite blocks, 20 × 20 × 40 cm, density 1.65 Outside dimensions of reflector 1.2 × 2 × 2 m; nominal thickness 40 cm Central region may be removed to form cavity for experiments	48. Shielding	Sides: 1.25 m heavy concrete blocks, density 3.5 Bottom: 35 cm concrete Top: 50 cm concrete lid plus 50 cm additional concrete slabs during operation
47. Radiation levels	2.5 mrem/hr		

CONTAINMENT

49. Type and material	No actual containment The reactor is located in a semi-airtight concrete building	50. Surroundings	Mainly farmland, sparsely populated, 1 km from nearest town with about 12000 inhabitants

COST ESTIMATE

51. Reactor and building	Reactor 300000 DM Building 750000 DM	53. Operating costs	300000 DM per year plus 300000 DM per year for depreciation
52. Support facilities	Not available	54. Staff requirements	4 physicists 3 laboratory personnel 2 mechanics

REATTORE ARGONAUT AGIP-NUCLEARE

PURPOSE: Research

DATE OF INFORMATION: November 1964

GENERAL

1. Reactor type	Argonaut type, medium enriched (20%) uranium, light water moderated and cooled, graphite and light water reflected	5. Owner and operator	AGIP Nucleare SNAM — LABORATORI
2. Nominal reactor power	10 kW (th)	6. Designer and builder	AGIP Nucleare SNAM LABORATORI
3. Purpose	Reactor physics, isotope production, educational purposes	7. Present status & construction schedule	In operation Reactor critical: May 1963
4. Location	Laboratorio di Ingegneria Nucleare de Montecuccolino, Bologna		

REACTOR PHYSICS

8. Neutron energy and lifetime	Thermal Lifetime 2×10^{-4} sec	10. Neutron flux	Thermal max. 10^{12} n/cm² sec
9. Core parameters	$\eta = 2.06$ $\varepsilon = 1$ $f = 0.81$ $p = 1$ $k_{\infty} = 1.69$ $k_{eff} = 1.002$ Fast leakage factor 0.612 Thermal leakage factor 0.969	11. Reactivity balance	Max. built in 0.5% $\frac{\Delta k}{k}$

CORE

12. Shape and dimensions	Annular 1.2 m high, 0.9 m o.d., 0.6 m i.d.	18. Average power density in core	0.1 to 0.25 kW/litre
		19. Burnup	Not available
13. No. of channels & subassemblies	24 boxes, each having 15 to 19 fuel plates	20. Fuel loading and unloading	Manually
14. Lattice	24 positions spaced equally on a circle of a radius of 37.5 cm	21. Irradiated fuel storage	Fuel storage pit, with heavy concrete plugs
15. Critical mass	2 to 4.5 kg ^{235}U according to core configuration	22. Moderator	500–800 litres light water av. temp. 20° C, max. temp. 100° C, forced cooling
16. Core loading at rated power	2 to 4.5 kg ^{235}U		
17. Average specific power in fuel	2.2 to 5 kW/kg ^{235}U	23. Blanket gas	None

FUEL ELEMENT

24. Form and composition	Rectangular plates 610 × 72 × 2.44 mm; enrichment 20% U_3O_8 in aluminium matrix	25. Cladding	Aluminium, 0.5 mm thick
		26. Subassemblies	A fuel subassembly consists of 15 to 19 plates

SIEMENS ARGONAUT REACTOR GRAZ

PURPOSE: Research and Training

DATE OF INFORMATION: October 1964

GENERAL

1. **Reactor type**	Argonaut type, highly enriched (20 %) uranium, light water moderated and cooled, graphite reflected	5. **Owner and operator**	Verein zur Förderung der Anwendung der Kernenergie, Graz
2. **Nominal reactor power**	1 kW	6. **Designer and builder**	Siemens Schuckertwerke A.G., Erlangen, Germany
3. **Purpose**	Materials test, reactor and nuclear physics, training	7. **Present status & construction schedule**	In operation Start of construction: Spring 1962 Reactor critical: May 1965
4. **Location**	Reactor Institut, Graz		

The reactor is similar to the Argonaut reactor at Argonne National Laboratory, USA, described in this Directory Vol. V, p. 111, in its main parts, with the following modifications:

Neutron flux: $6.3 \cdot 10^{11}$ n/cm² sec

Av. specific power in fuel: 0.5 kW/kg ^{235}U for one slot core
0.227 kW/kg ^{235}U for annular core

Irradiated fuel storage: 16 storage holes in concrete

Control rods: 2 shim-safety cadmium plates 220 × 250 × 0.76 mm
2 regulating cadmium plates 220 × 250 × 0.76 mm
Worth of shim-safety and regulating plates
2.3 to 4.7 $\frac{\Delta k}{k}$ depending on core configuration

RESEARCH FACILITIES

Designation	No	Position	Useful dimensions (cm)	Neutron flux (n/cm²sec)	Remarks
Horizontal neutron beam reaching core	6	(1)	2 channels 6 × 8 4 channels 10 × 10	th. $3.61 \cdot 10^{7}$ f. $0.8 \cdot 10^{7}$	at 960 W
Horizontal neutron beam reaching reflector	11	(2)	10 × 10	th. $1.2 \cdot 10^{7}$ f. $0.17 \cdot 10^{7}$	at 960 W
Vertical neutron beam reaching reflector	1	(3)	10 diam.	th. $1.4 \cdot 10^{7}$	at 1 kW
Thermal column	1	(4)	156 × 153.2 × 123.7		Graphite 10—100
Shield test	1	(5)	218.5 × 122 × 110		Water filled tank

Exhibit S

HAZARDS ANALYSIS BY THE TEST & POWER REACTOR SAFETY BRANCH

DIVISION OF LICENSING AND REGULATION

IN THE MATTER OF

THE UNIVERSITY OF CALIFORNIA (UCLA)

DOCKET NO. 50-142

By application amendments dated February 23, 1961 and April 15, 1961, The University of California (UCLA) has requested AEC authorization to make several minor modifications to their 10 Kw training reactor. These modifications are discussed below:

Discussion of Proposed Modifications

Low Temperature Inhibit Condition

Rod withdrawal is currently inhibited if the moderator temperature is below 80°F; this condition was considered necessary at the design stage of the UCLA reactor as calculations indicated a positive temperature coefficient below this point. Recent measurements have indicated that the coefficient is actually negative below this temperature (-8.5×10^{-5} delta k per centigrade degree over the range 32°F to 120°F). The applicant has therefore proposed to eliminate this inhibit condition.

Short Period Inhibit Condition

Rod withdrawal is currently inhibited in the event of a reactor period less than 10 seconds. The applicant proposes to change this set point from 10 seconds to 6 seconds in order to avoid spurious period indications that have become an operational inconvenience. This modification will not affect the period scram set point which will remain at a setting of 3 seconds.

Low Count Rate Inhibit

Rod withdrawal is currently inhibited if the neutron count level is below 10 counts per second. The applicant proposes to change this inhibit condition to 1.5 counts per second in order to allow withdrawal of the BF_3 startup counter to a lower flux region. The value of 1.5 counts per second is consistent with values specified for other research reactors; e.g., the University of Florida reactor, an essentially similar type, requires a minimum of 2 counts per second.

Change of Source

The applicant proposes to replace the 2 curie Pu-Be source with a 10 millicurie Ra-Be source. The present Pu-Be source has been determined to give a much stronger indication than required for safe startup. Both types of sources have been utilized successfully in research reactors; we anticipate that no additional hazard will result from the replacement of Pu-Be source with the Ra-Be source.

We have examined the safety aspects of each of the modifications described above. In our opinion, these modifications all represent minor changes and will not adversely affect the safe operation of the UCLA reactor.

By further application amendment dated March 21, 1961, the University has requested authorization to perform a number of experiments. The Staff considered it advisable at the time of issuance of the operating license to limit the performance of experiments as there were essentially no experimental procedures described in the original application, and proposed experiments other than those in the initial testing program were not specifically described. We believe that sufficient information now has been submitted in support of the experiments requested by the March 21 application to enable us to make a safety evaluation. The experiments requested by this application are described below:

Discussion of Proposed Experiments

Neutron Beam Experiments

These experiments will allow the applicant to extract a beam of neutrons from the reactor core upon removal of shield plugs and stringers associated with the beam ports, access holes, or thermal column. Adequate shielding will prevent over exposure of personnel. Experiments of this type are very common for University research reactors.

Operation With Shield Blocks Removed

The applicant proposes to operate at power levels below one watt with the portable central shield blocks removed and to bypass the interlock which inhibits withdrawal of the control rods if the reactor closures are not in place. Radiation levels at power levels below one watt will be sufficiently low to allow removal of the blocks; radiation surveys and personnel monitoring will be employed to detect any possible radiation hazards.

Irradiation of Special Nuclear or Source Materials

The applicant proposes to irradiate up to 250 grams of fissionable material in double sealed containers for periods less than four hours at full power. This material will be irradiated at the outer face of the thermal column. Calculations and experiments indicate that the introduction of fissionable materials at this point (48 inches from the core fuel) will not consitute an increase of reactivity for the reactor itself. The limitation to less than 250 grams insures that the material irradiated cannot assume a critical configuration. We concur with the applicant as to the reactivity coupling and criticality effects; we believe no safety problems will be created by the irradiation of this material.

Control Rod Positions

The applicant proposes to operate the reactor with the three safety rods partially inserted to varying depths so as to make flux distribution studies possible. The regulating rod will be on its down limit during any manipulation of a safety rod and the reactor will be brought to criticality only through withdrawal of the regulating rod. The experiment will not involve any change in the allowable excess reactivity (0.65 delta k/k).

Irradiation of Other Material

The applicant proposes to irradiate absorbing material in the core, reflector, shield, thermal column, or shield tank of the reactor. Only material resulting in negative reactivity will be involved and no change of fuel loading will be permitted. Since the reactor will be loaded to a maximum of 0.6% delta k/k, no serious insertions of reactivity would result upon inadvertent removal or failure of experiments. Irradiation of absorbing material is very common in University research reactors.

Reactor Oscillation

The applicant proposes to perform reactor oscillation experiments, utilizing a rotor-stator type pile oscillator. No changes in fuel loading will be permitted and the magnitude of the oscillation will be adjusted so as to result in a power variation of no more than plus or minus five per cent. Similar experiments have been performed in other research reactors; we anticipate no new safety problems will result from the use of the pile oscillator in the UCLA reactor.

Water Level Variations

The applicant proposes to operate the reactor at powers up to one watt with the reactor core water below its normal operating level and with zero coolant flow. To accomplish these experiments, the core water level, primary coolant pump, and water flow safety interlocks will be bypassed. The written procedures employed during normal startup will be followed at each new water level; the rods will be reinserted before any change in water level is made. Reactivity effects will be negative upon lowering the water level and in no experiment will the excess reactivity be permitted to exceed 0.6% delta k/k. Undesirable temperature rises will be avoided by operating the reactor at low power levels; an alarm light and an alarm horn will be actuated in event of a high moderator temperature. It is our opinion that these experiments can be performed as proposed without presenting any hazard.

Temperature Variations

The applicant proposes to perform experiments involving reactivity changes induced by varying the core coolant temperature over the range 33°F to 120°F. Reactivity changes will be introduced by successively bypassing the primary pump and coolant flow safety interlocks, stopping the primary coolant flow, establishing a different coolant temperature, and then restoring the coolant flow. The maximum reactivity insertion possible is about 0.4% delta k/k, which corresponds to a period of about 5 seconds. The 3 second scram will be operative during these experiments. The coolant flow will be interrupted only when the reactor is at power levels below one watt. We believe that experiments of this type can be conducted safely in the manner proposed.

It is pertinent to note that the experiments described above will be conducted under the direct supervision of the Reactor Supervisor or his licensed deputy in accordance with written procedures approved by the UCLA Reactor Hazards Committee. We are satisfied that performance of the proposed experiments will not present undue hazard to the public or operating personnel.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

THE REGENTS OF THE UNIVERSITY OF CALIFORNIA

DOCKET NO. 50-142

LICENSE

License No. R-71

1. This license applies to the Argonaut-type nuclear reactor (hereinafter referred to as "the reactor") designed for 10 kilowatt (thermal) operation which is owned by The Regents of The University of California and located on the University of California campus in Los Angeles, California, and described in the application dated June 24, 1959, and amendments thereto dated January 4, 1960, and June 23, 1960, (hereinafter collectively referred to as "the application").

2. Pursuant to the Atomic Energy Act of 1954, as amended, (hereinafter referred to as "the Act") and having considered the record in this matter, the Atomic Energy Commission (hereinafter referred to as "the Commission") finds that:

   A. The reactor has been constructed in conformity with Construction Permit No. CPRR-42 issued to The Regents of The University of California and will operate in conformity with the application and in conformity with the Act and with the rules and regulations of the Commission;

   B. There is reasonable assurance that the reactor can be operated at the designated location without endangering the health and safety of the public;

   C. University of California is technically and financially qualified to operate the reactor, to assume financial responsibility for payment of Commission charges for special nuclear material and to undertake and carry out the proposed use of such material for a reasonable period of time, and to engage in the proposed activities in accordance with the Commission's regulations;

   D. The possession and operation of the reactor and the receipt, possession and use of the special nuclear material in the manner proposed in the application will not be inimical to the common defense and security or to the health and safety of the public; and

APPENDIX "A"

TO

UNIVERSITY OF CALIFORNIA

FACILITY LICENSE NO. R-

Estimated Schedule of Transfers of Special Nuclear Material from the Commission to the University and to the Commission from the University:

(1)	(2)	(3)		(4)	(5)
Date of Transfer (Fiscal Year)	Transfers from AEC to the University Kgs. U-235	Returns by the University to AEC Kgs. U-235 Recoverable Cold Scrap	Spent Hot Fuel	Net Yearly Distribution Including Cumulative Losses Kgs. U-235	Cumulative Distribution Including Cumulative Losses Kgs. U-235
1960	4.000	0.660	-	3.340	3.340
1961	-	-	-	-	3.340
1962	-	-	-	-	3.340
1963	-	-	-	-	3.340
1964	-	-	-	-	3.340
1965	0.010	-	0.005	0.005	3.345
1966	-	-	-	-	3.345
1967	-	-	-	-	3.345
1968	-	-	-	-	3.345
1969	-	-	-	-	3.345
1970	0.010	-	0.005	0.005	3.350
1971	____	____	3.340*	(3.340)	.010**
	4.020	0.660	3.350	0.010**	

* Inventory to be returned

** Burnup losses

APPLICATION FOR A CONSTRUCTION PERMIT
FOR A TRAINING REACTOR FACILITY

Date of Application: May 30, 1959

To: U. S. Atomic Energy Commission
1901 Constitution Avenue
Washington 25, D. C.

Attention: Division of Civilian Application

Application based on: Code of Federal Regulations
Title 10, Part 50

Section 50.33

a. Name of Applicant: College of Engineering
University of California
at Los Angeles

b. Address of Applicant: Los Angeles 24, California

c. Description of business or occupation of applicant: Education and research in all branches of engineering.

d. (1) and (2) not applicable.
(3) (i) State where organized: California
Principal location of business: Los Angeles 24, California

(ii) Names, addresses, and citizenship of principal officers:

Name	Title	Address	Citizenship
Clark Kerr	President	Berkeley, Cal.	U. S.
D. S. Allen	Chancellor Los Angeles Campus	Los Angeles, Cal.	U. S.
L. M. K. Boelter	Dean, College of Engineering	Los Angeles, Cal.	U. S.
T. T. Hicks	Associate Professor	Los Angeles, Cal.	U. S.

(iii) Foreign Relationships: The applicant is in no way owned, controlled, or dominated by an alien, a foreign corporation, or foreign government.

(4) Agent: The applicant is not acting as the agent or representative of another person in filing this application. The applicant is the principal party.

e. Class of license applied for:

Construction Permit only. In August of this year we will ask that the Construction Permit be converted to a Class 104 License.

Use to which the facility will be put:

The reactor and its supporting laboratories will be used for the training and education of senior undergraduate and graduate students in nuclear engineering and related sciences. In addition to formal courses and demonstrations, the reactor will be used to support research at the Master's and Ph.D. level.

Period of time for which license is requested:

At the time the Class 104 License is applied for, in a supplement to this document, we will ask that the license run for ten (10) years, or until November 30, 1969.

Other licenses applied for in connection with this facility:

Special Nuclear Material - (1) 4.00 Kg 90% enriched U-235; (2) Pu-239 as 2 curie Pu-Be neutron source. An allocation of Special Nuclear Material is requested in this application.

f. Financial qualifications of the applicant:

The College of Engineering is a part of the University of California, Los Angeles, which is part of the combined University of California -- a state university and land grant college. Its financial support is primarily from appropriations of the California State Legislature. Additional income is derived from fees, grants, and contracts. The fiscal 1959 budget of the University is approximately $200,000,000, of which approximately $2,000,000 is budgeted for the College of Engineering, Los Angeles. The budget request for operation of the reactor facility in 1960-61 is $97,000. Sufficient funds are available to operate the reactor facility on a continuing basis for the duration of its license.*

* Note: See Item e of this document.

under this application and future supplements. The following information is provided as required under this section.

1. The applicants financial qualifications are discussed in Section 50.33 paragraph f.

2, 3, and 4. Estimated date for receipt of first shipment of special nuclear material: It is desired that 4 Kg of U-235 as 90% enriched fuel be received in Los Angeles, California approximately October 1, 1959. Approximately 660 grams contingency allowance will be returned to AEC on or about December 30, 1959.

Prior to shipment to Los Angeles this fuel is to be fabricated into fuel plates as specified by AMF Atomics, the reactor fabricator. An allowance of 10% excess fuel should allocated for waste in fabrication.

It is desired that the 2 curie Pu-Be source containing 30 gms. of Pu-239 be received in Los Angeles on or about August 15, 1959.

Estimated schedule by years for subsequent receipts, consumption and transfer:

Year	Receipt by UCLA	Consumption	Transfer to Commission	Pu Production
1960	None	0.975 gm. 235	None	0.05 gm.
1961	None	0.975 gm. 235	None	0.05 gm.
1962	None	0.975 gm. 235	None	0.05 gm.
1963	None	0.975 gm. 235	None	0.05 gm.
1964	10 gm. 235	0.975 gm. 235	5gm 235 07mg Pu	0.05 gm.
1965	None	0.975 gm. 235	None	0.05 gm.
1966	None	0.975 gm. 235	None	0.05 gm.
1967	None	0.975 gm. 235	None	0.05 gm.
1968	None	0.975 gm. 235	None	0.05 gm.
1969	10 gm. 235	0.975 gm. 235	5gm 235 07mg Pu	0.05 gm.

This schedule is based on the assumption that reactivity will be replenished by adding one replacement fuel plate containing 10 gm U-235 at five-year intervals.

It is assumed that there will be no return of the entire core to the Commission due to corrosion or obsolescence during the ten year period for which this schedule applies.

Since no fabrication or reprocessing will be done by the university, there will be no operating losses of special nuclear material.

Supporting date for above estimates:

a. Fuel loading for UCLA training reactor
Fuel enrichment 90%

Calculated cold clean critical mass $\frac{\text{gm U-235}}{3200}$

Exhibit T

Exhibit T
16 pages of SNM license correspondence

UNIVERSITY OF CALIFORNIA, LOS ANGELES

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SCHOOL OF ENGINEERING AND APPLIED SCIENCE
LOS ANGELES, CALIFORNIA 90024

"P"

3 June 1970

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Morris:

We wish to request permission to have on site in the Nuclear Energy Laboratory an additional 4.3 Kg of Uranium-235 for the purpose of refueling. Money has been granted us by the Division of Nuclear Education and Training for a new fuel loading consisting of 24 bundles for a complete core change and 5 spare bundles for low power experimentation. Our current license permits us to have only 3.5 Kg of U-235 in the Nuclear Energy Laboratory at any one time. However, during the actual refueling, we would have approximately 7.8 Kg of U-235 and after refueling and shipment of the old fuel bundles approximately 4.3 Kg.

Our present plans are to refuel the reactor during the summer of 1971.

Yours truly,

Thomas E. Hicks, Director
Nuclear Energy Laboratory

TEH:ch

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USAEC
JUN 8 1970
REGULATORY
MAIL SECTION
DOCKET CLERK

RECEIVED
JUN 8 1970
U.S. ATOMIC ENERGY COMMISSION
DIVISION OF REACTOR LICENSING

Fourth Cert

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Docket File
D. J. Skovholt
R. H. Vollmer
F. Schroeder
H. K. Shapar, OGC
D. L. Ziemann
E. R. Fleury
R. Diggs
PDR

"P"

JUN 24 1970

Docket No. 50-142

University of California at
Los Angeles
School of Engineering and
Applied Science
Los Angeles, California 90024

Attention: Dr. Thomas E. Hicks, Director
Nuclear Energy Laboratory

Gentlemen:

Your letter dated June 3, 1970, requested an increase of 4.3 kilograms (from 3.5 kilograms to 7.8 kilograms) in the quantity of uranium 235 which the University may possess at any one time under Facility License No. R-71, to accommodate the refueling of your Argonaut-type research reactor which is planned for the summer of 1971. However, the license currently authorizes the University to receive, possess, and use 4 kilograms of uranium 235 which, when increased by 4.3 kilograms, is a total of 8.3. Please advise us whether you wish the new limit to be 8.3 kilograms or 7.8 kilograms.

If you do not plan to receive the additional quantity of material for refueling prior to the summer of 1971, we will take action on this request at the same time we consider your application dated February 20, 1970, for renewal of License No. R-71 and the incorporation of Technical Specifications. Please indicate your plans in this regard.

Your request will necessitate an amendment to your license. Please note that Section 50.30(b) of 10 CFR Part 50 requires that applications for amendments be filed under oath or affirmation with three signed originals and nineteen additional copies. Therefore, your response to this letter should be filed in accordance with Section 50.30(b) and incorporate by reference the letter dated June 3, 1970.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing

Circulated 6/25/70

OFFICE ▶	DRL	DRL	DRL	DRL	DRL	DRL
SURNAME ▶	RDiggs:pdl ERFleury	JShapaker	DLZiemann	DJSkovholt	FSchroeder	PAMorris
DATE ▶	6/22/70	6/23/70	6/23/70	6/24/70	6/24/70	6/24/70

50-142

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SCHOOL OF ENGINEERING AND APPLIED SCIENCE
LOS ANGELES, CALIFORNIA 90024

"P"

9 July 1970

Mr. Donald J. Skovholt
Assistant Director of Reactor Operations
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Skovholt:

In reference to your letter of 24 June 1970, the amount of additional fuel we expect to receive will be 4.3 kilograms. Therefore we would like the new limit to be 8.3 kilograms of uranium-235.

Our present plans call for refueling in June of 1971. We would prefer to have the renewal of License R-71 approved rather than writing an amendment to our current license. The proposed license which we sent with our Technical Specifications includes the provision of up to 6 kilograms of fuel in storage for refueling (paragraph 3B).

Yours truly,

Thomas E. Hicks

Thomas E. Hicks, Director
Nuclear Energy Laboratory

TEH:ch

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SCHOOL OF ENGINEERING AND APPLIED SCIENCE
LOS ANGELES, CALIFORNIA 90024

10 September 1970

R-24-70

Dr. Peter A. Morris
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Morris:

Previous requests for permission to possess additional 93% enriched U-235 in the form of a new fuel loading for our reactor were based on the proposed final composition of the fuel elements.

We have now been advised that in order to process these fuel elements, it will be necessary to provide the manufacturer with 6 kilograms of melt stock. Any material remaining (over and above that contained in the finished plates, less an estimated 2% loss during manufacture) would be delivered to UCLA with the fuel elements.

Based on the above we need to possess a total of 10 kilograms of 93% enriched U-235 (4.0 kilograms now covered by our R-71 license plus 6.0 kilograms required to fabricate our new loading).

Our request for Technical Specifications contains provision for possession of 10 kilograms of 93% enriched U-235. However, delays in negotiating these Technical Specifications make it necessary that we now ask for an amendment to our existing license in order to expedite fabrication of these fuel elements.

NUCLEAR ENERGY LABORATORY

THOMAS E. HICKS, Professor and Director

Enclosed are three official copies of the Application for Amendment to the Facility License R-71 and nineteen additional copies. If there are any questions regarding this request, please contact Mr. D. N. Jones, Laboratory Manager (213) 825-2187.

Very truly yours,

Thomas E. Hicks

Thomas E. Hicks
Director, Nuclear Energy Laboratory

TEH:cn
Enclosure: 22 copies of application

3-24-70

APPLICATION FOR AMENDMENT

to the

Facility License (R-71)

for the

UCLA Training Reactor

Amendment Request Number 8

10 September 1970

We wish to amend sub-paragraph 3B to read:

3B. Pursuant to the Act and Title 10 CFR Chapter I, Part 70, Special Nuclear Material, to receive, possess and use up to 4 kilograms of contained U-235, 32 grams of plutonium in a Pu-Be source, and one gram of plutonium in the form of foils or wires for the purpose of making flux distribution measurements, plus 6 kilograms of 93% U-235, required for fabrication of a new fuel loading (24 bundles and 5 spare bundles; a total of 319 plates assembled), all for use in connection with the reactor; . . . .

Note: Underlined portion represents the change requested.

DOCKET NO. 50-142

UNIVERSITY OF CALIFORNIA, LOS ANGELES

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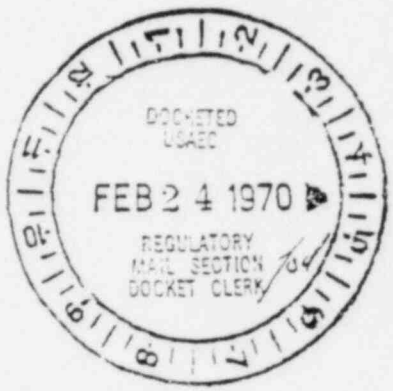
OFFICE OF THE CHANCELLOR
LOS ANGELES, CALIFORNIA 90024

February 20, 1970

"P"



Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Skovholt:

Pursuant to your letter of 21 January 1970, attached are 22 copies of the Application for Renewal of the Reactor Facility License No. R-71 at the University of California, Los Angeles. This application is of the technical specifications type.

1. The ultimate responsibility is vested in the Regents of the University of California.

2. The immediate administrative authority is vested in Charles E. Young, Chancellor, University of California Los Angeles, Los Angeles, California 9002-.

3. The contact point at UCLA for all correspondance, license amendments, and inspection reports is Mr. Harold W. Brown, Environmental Health and Safety Officer, University of California Los Angeles, Los Angeles, California 90024.

4. Direct administration rests in the Office of Environmental Health and Safety.

5. Semi-annual special nuclear material inventory reports will be supplied by the Office of Environmental Health and Safety.

6. Conditions of 10 CFR will be complied with.

7. All internal regulations of the University will be complied with.

8. The proposal has been approved by the Engineering Radiation Use Committee and by the Campus-Wide UCLA Radiation Safety Committee, which must approve any significant changes in operating procedures of the facility.

Yours truly,

David S. Saxon
The Vice Chancellor

HWB:vs
Enclosure

Proposed License

UCLA-1 Research Reactor

1. Applicability:

This license supersedes and replaces License R-71 and its amendments and changes. It is applicable to the UCLA Argonaut -type research reactor (hereinafter referred to as "UCLA-1") which is owned by the Regents of the University of California, and is located on the campus at Los Angeles, California, (hereinafter called "the University").

2. Demonstrated Performance:

By its record in constructing the reactor and operating it safely for a period of 10 years, the University has demonstrated its technical and financial qualifications to operate the reactor in accordance with the United States Atomic Energy Commission's regulations, (hereinafter called the "Commission"), and in a manner consistent with the health and safety of the public.

Therefore, for the purpose of this license renewal, the University's operating record will be accepted as satisfying the requirements of section 50.34 of Part 50, of the Commission's regulations, except for the restrictions spelled out in this license.

3. License:

Subject to the conditions and requirements incorporated herein, the University hereby requests the Commission to license it as follows:

A. Pursuant to Section 104c of the Atomic Energy Act of 1954 and Title 10, CFR, Chapter 1, Part 50, "Licensing of Production and Utilization Facilities", to possess and operate UCLA-1 as a utilization facility at the designated location in Los Angeles, California, in accordance with the procedures and limitations described in the application and this license;

B. Pursuant to the Act and Title 10, CFR, Chapter 1, Part 70, "Special Nuclear Material", to receive, possess and use up 10.0 kilograms (4.0 kg in the reactor and up to 6.0 kg of fuel in storage for refueling) of uranium 235, 500 grams of plutonium, and 250 grams of uranium 233 for use in connection with operation of the reactor or other research projects;

C. Pursuant to the Act and Title 10, CFR, Chapter 1, Part 30, Licensing of Byproduct Material", to possess but not to separate such byproduct materials as may be produced by operation of the reactor.

This license shall be deemed to contain and be subject to the conditions specified in Part 20; in Sections 50.54 and 50.59 of Part 50 and Section 70.32 of Part 70, Title 10, Chapter 1, CFR, and to be subject to all applicable provisions of the Act, and to the rules and regulations and orders of the Commission, now or hereafter in effect, and to the additional conditions specified below:

4. Operating Restrictions:

A. Maximum Power Level

The University is authorized to operate UCLA-1 at power levels (steady-state or transient) up to 500 kilowatts thermal, so long as the operating and safety limits described in the Technical Specifications (Appendix A) are not exceeded.

B. Technical Specifications

The Technical Specifications for operating UCLA-1 are contained in Appendix A. They are part of this license. The University shall operate the facility in accordance with the Technical Specifications and may make changes therein only when authorized by the Commission in accordance with the provisions of section 50.59 of Title 10, CFR.

C. Records and Reports

The University shall maintain records as described in the Technical Specifications.

D. Term

This license is effective as of the date of issuance and shall expire at midnight, March 30, 1980.

AUG 31 1970

Docket No. 50-142

The Regents of the University
of California
ATTN: Mr. H. V. Brown
Environmental Health and
Safety Officer
Department of Engineering
Los Angeles, California 90024

Gentlemen:

Your application of February 20, 1970, requested renewal of Facility License No. R-71, authorization to increase the steady state power level of the UCLA-1 research reactor from 100 kWt to 500 kWt, and an increase in the quantities and kinds of special nuclear material that you may receive, possess and use in connection with the operation of the reactor. Proposed Technical Specifications for operation of your Argonaut-type research reactor were also submitted.

On the basis of our preliminary review of your application, we find that we need the following additional information to complete our evaluation:

1. A description of the physical form of the 250 grams of uranium 233 and the additional 467 grams of plutonium requested and a description of the proposed use of this material.

2. A supplemental safety analysis report, as described in Section 50.34 of 10 CFR Part 50, in support of your request to increase the power level to 500 kWt.

3. A revision of the format and content of the proposed Technical Specifications to comply with the requirements of Section 50.36 of 10 CFR Part 50. In this regard, copies of the Technical Specifications for the University of Florida Argonaut-type research reactor and the Army Materials and Mechanics Research Center pool-type reactor are enclosed to provide additional guidance as to the format and content desired. Any changes in the facility or in the operating limitations as described in the currently approved Safety Analysis Report, as amended, that are incorporated into your proposed Technical Specifications must be described and evaluated in the supplemental safety analysis report.

OFFICE ▶

The above information should be filed as an amendment to your application with three signed and notarized original copies and nineteen additional copies. In addition, your letter of July 9, 1970, which requests an increase in the quantity of uranium 235 which the University may possess under Facility License No. R-71 to 8.3 kilograms, should be incorporated in the amendment by reference.

If you desire further information or clarification of these requests, please contact Messrs. Dennis L. Ziemann or James W. Shapaker.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director for
Reactor Operations
Division of Reactor Licensing

Enclosures:
1. 10 CFR Part 50
2. Army Materials and Mechanics Research Center Technical Specifications
3. University of Florida Technical Specifications

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OFFICE ▶	DRL	DRL	DRL	DRL	DRL	
SURNAME ▶	RHVollmer:bsp	ERFleury	JWShapaker	DLZiemann	DJSkovholt	
DATE ▶	8/27/70	8/26/70	8/28/70	8/28/70	8/31/70	

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SCHOOL OF ENGINEERING AND APPLIED SCIENCE
LOS ANGELES, CALIFORNIA 90024

7 October 1970

Dr. Peter A. Morris
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Docket No. 50-142
Regulatory File Cy.

Dear Dr. Morris:

Enclosed is a revised application for renewal of license R-71. In accordance with the letter received from Donald J. Skovholt, Assistant Director for Reactor Operations, dated 31 August 1970, we have revised our license and technical specifications to include the additional information required.

Our request for 250 grams of uranium 233 has been deleted. The request for 467 grams of plutonium has been revised to 33 grams of plutonium, as outlined in the license. We have not included the supplemental safety analysis suggested in the letter of 31 August 1970, as we have omitted the request to increase the power level to 500 kilowatts.

The technical specifications (Appendix A) have been revised to conform to the form and content of the technical specifications prepared for the University of Florida Argonaut-type research reactor.

The license and technical specifications have been reviewed and approved by the Radiation Use Committee and the Radiation Safety Committee.

NUCLEAR ENERGY LABORATORY

THOMAS E. HICKS, Professor and Director

We would appreciate your prompt attention to our request for approval of the enclosed license and technical specifications and look forward to your reply.

If there are further questions, please contact Mr. D. N. Jones, Laboratory Manager, (213) 825-2187.

Very truly yours,

Thomas E. Hicks

Thomas E. Hicks, Director
Nuclear Energy Laboratory

TEH:ch
Enclosure: license application

EXHIBIT 6

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-142

NOV 18 1974

The Regents of the University
of California
Nuclear Energy Laboratory
ATTN: Mr. Thomas E. Hicks
Director
Los Angles, California

Gentlemen:

Your letter of October 28, 1974 stated that UCLA was considering methods to reduce their Special Nuclear Material inventory below the formula quantity specified in Title 10, Code of Federal Regulations, Part 73.

As of this date we have not received a written confirmation that you have reduced your Special Nuclear Material inventory nor have we received a request to review your security plan assuming the inventory was reduced. You are reminded that your original plan, as submitted, was not acceptable and that you may be in violation of Title 10, Code of Federal Regulations, Part 73. Noncompliance with the Regulations would require that appropriate enforcement action be taken by us.

Your response is requested within seven days of the receipt of this letter.

Sincerely,

for George Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing

G. L. Rogosa

UNIVERSITY OF CALIFORNIA, LOS ANGELES

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SCHOOL OF ENGINEERING AND APPLIED SCIENCE
LOS ANGELES, CALIFORNIA 90024

Boelter Hall 2567
March 1, 1979

Mr. C. A. Berger, Contracts Branch
U.S. Department of Energy
San Francisco Operations Office
1333 Broadway
Oakland, California 94612

Re: Contract EY-76-03-034, P.A. 192

Dear Mr. Berger:

By copy of our letter of November 9, 1978 to Dr. Rogosa; you were advised of our request to DOE for support of the cost of shipping some excess irradiated fuel to the Idaho Chemical Reprocessing Plant. The estimated cost of the operation is approximately $4000, and support was sought under the subject contract.

Mr. D. G. McIntosh (DOE/SAN) has been helpful in arranging for the physical transfer and shipment. These plans are going forward.

Paragraph 3 of our letter to Dr. Rogosa outlined the basis of our request. We have not yet received a response. We are presently in technical violation of our SNM possession limit, and further delay could invite a Notice of Violation by the Nuclear Regulatory Commission. Your immediate action is now requested. Please call us if you have any questions.

Sincerely,

Hardy Dhillon
Contract and Grant Officer
(213) 825-0695

Ivan Catton, Professor and Director
Nuclear Energy Laboratory
(213) 825-2040

IC/li

cc: D. G. McIntosh, DOE/SAN
✓G. L. Rogosa, DOE, Division of Nuclear Physics
R. R. O'Neill, Dean, UCLA/SEAS
C. E. Ashbaugh, UCLA/SEAS/NEL
R. H. Engelken, USNRC, Region V

Exhibit U

IAEA-TECDOC-233

RESEARCH REACTOR CORE CONVERSION FROM THE USE OF HIGHLY ENRICHED URANIUM TO THE USE OF LOW ENRICHED URANIUM FUELS GUIDEBOOK

PREPARED BY A CONSULTANTS' GROUP,
COORDINATED AND EDITED BY THE
PHYSICS SECTION
INTERNATIONAL ATOMIC ENERGY AGENCY

A TECHNICAL DOCUMENT ISSUED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1980

CONTRIBUTING ORGANIZATIONS

Argonne National Laboratory	ANL	United States of America
Atomic Energy Research Establishment Harwell	AERE Harwell	United Kingdom
Comisión Nacional de Energía Atómica	CNEA	Argentina
Commissariat a l'Énergie Atomique	CEA	France
Compagnie Pour l'Étude et la Réalisation de Combustibles Atomiques	CERCA	France
Eidg. Institut für Reaktorforschung	EIR	Switzerland
General Atomic Company	GA	United States of America
General Electric Company Reactor Equipment Ltd.	GEC	United Kingdom
Internationale Atomreaktorbau GmbH	INTERATOM	Federal Republic of Germany
Japan Atomic Energy Research Institute	JAERI	Japan
Kernforschungszentrum Karlsruhe GmbH	KFK	Federal Republic of Germany
Kyoto University Research Reactor Institute	KURRI	Japan
NUKEM GmbH	NUKEM	Federal Republic of Germany
Österreichische Studiengesellschaft für Atomenergie	ÖSGAE	Austria

The IAEA is grateful for the contributions volunteered by these organizations and thanks their experts for preparing the detailed investigations and for evaluating and summarizing the results presented in this Guidebook.

SUMMARY

This Guidebook has been prepared to assist reactor operators and physicists in determining both the feasibility of converting their specific reactors from HEU to LEU fuel and the options available for implementation. A wide variety of information is presented on the physics, thermal-hydraulics, and fuels of light water moderated and cooled research and test reactors. Most of the methods discussed in this Guidebook can also be directly applied to the analysis of research reactors containing heavy water as moderator and/or coolant. However, in consideration of the special features of heavy water reactors, an addendum to this Guidebook is planned to address the feasibility of converting these reactors to LEU fuel and the options available for implementation.

The following is a brief outline of how the results were obtained, and how this Guidebook can be used most effectively.

1. Actions Needed For Conversion From HEU* Fuels to LEU* Fuels

Section 1.5 gives a summary of the type of studies that are needed to prepare for core conversion.

It is possible for these studies to be performed by the reactor operators/physicists themselves, or with the aid of laboratories which have offered technical assistance. Appendix G lists the typical data needed for enrichment reduction conversion studies. Section 1.4.2, Chapter 3, and Appendix H contain information on the current status, development potential, and commercial availability of fuels with high uranium densities. Appendix I analyses the main economic aspects of core conversions to LEU fuel.

2. Generic Studies

Calculations have been performed by different laboratories for two generic MTR-type reactors with power levels of 2 MW and 10 MW to determine their potential for conversion. The results are summarized in Section 2 and include the uranium densities that would be required with different fuels and fuel element designs, the corresponding thermal-hydraulic safety margins, and the performance that would be expected from the converted core. Detailed information on the methods and procedures used and the results obtained for the various core conversion options are presented in Appendices A through D.

3. Specific Studies

The methods and results of core conversion studies for two specific reactors with power levels of 3.5 MW and 50 MW, respectively, are provided in Appendix E.

4. Benchmark Calculations

In order to compare the accuracy of calculation methods used in the different research centers, benchmark problems were defined and calculated with the different methods. The main core calculations using 93%, 45% and 20% enrichment are based on an idealized 6 x 5 element, plate-type core with a power of 10 MW reflected by single graphite rows on two sides, and surrounded by water. Results of the calculations, including cross section data, and descriptions of various burnup conditions are summarized in Section 2.4 and described in detail in Appendix F. As a first step in core conversion, it is recommended that reactor operators/physicists use their own methods and codes to calculate this benchmark problem, and to compare the results.

5. IAEA Assistance

The IAEA can be contacted, through official channels, to provide assistance for the core conversion of specific reactors. The IAEA can offer coordinating assistance between reactor organizations and those laboratories in the USA, the FRG, and France which have offered technical assistance (Section 1.3). If necessary, the IAEA can also provide fellowships to visit those laboratories for joint studies on core conversions. The preparation of a second guidebook addressing safety and licensing issues related to core conversions is planned under the auspices of the IAEA.

*For simplicity, the following definitions have been adopted for this publication:

HEU - Highly Enriched Uranium ($\geq$70 wt% ^{235}U)

MEU - Medium Enriched Uranium (45 wt% ^{235}U)

LEU - Low Enriched Uranium (<20 wt% ^{235}U)

REU - Reduced Enriched Uranium (includes MEU and LEU)

CONTENTS

APPENDICES

1. MAJOR CONSIDERATIONS IN REACTOR CONVERSIONS

1.1 INTRODUCTION

In the 1950s and 1960s, low power research reactors were built around the world which utilized MTR-type fuel elements containing <20% enriched uranium (LEU). This value was chosen because it was considered to be a limit for weapon usable material. However, the demand for higher specific power created a need for greater ^{235}U concentrations and led to the substitution of highly enriched uranium (HEU) in place of the LEU fuel previously utilized. HEU also yielded other benefits including longer core residence time, higher specific reactivity, and somewhat lower cost. HEU then became readily available and was used for high power reactors as well as low power reactors where LEU would have sufficed. The trend toward higher and higher specific power also led to the development of the dispersion type fuels which utilized HEU with a density of about 1.6 - 1.7 g/cm^3.

In the 1970s, however, concerns were again raised about the proliferation-resistance of fuels and fuel cycles, and since enrichment reduction to less than 20% is internationally recognized to be a fully adequate isotopic barrier to weapons usability certain Member States have moved to minimize the international trade in highly enriched uranium and have established Reduced Enrichment Research and Test Reactor (RERTR) Programs. The goal of these programs is to develop the technical means, such as design modifications and development of new fuels, to assist in implementing reactor conversions to LEU fuels with minimum penalties. These programs have been established in the U.S., France, the Federal Republic of Germany, and Japan. It is anticipated that through the continued efforts of these programs, and with IAEA coordination, many reactors currently utilizing fuel element materials and designs less advanced than currently feasible may soon be converted to the use of LEU fuel. For other reactors, whose conversion to the use of LEU fuel may be feasible only after significant fuel development, a temporary decrease of the enrichment to an intermediate range of 45% (MEU) would be a worthwhile improvement in proliferation resistance.

Concern has also been expressed about the presence of plutonium in spent fuel, especially when the fuel is irradiated in reactors utilizing very low enrichment and/or operating at high powers, and it is necessary to consider both the plutonium produced and the enriched uranium in the overall assessment of the proliferation potential of a particular reactor.

1.2 REASONS FOR REACTOR CONVERSIONS TO LEU

Operators of research and test reactors that use highly enriched uranium may consider converting their reactors to the use of low enriched uranium fuels for several closely related reasons. One could be the desire to reduce the proliferation potential of research reactor fuels. A second reason could be a desire to increase the assurance of continued fuel availability in the face of probable restrictions on the supply of highly enriched uranium. A third reason could be the possible reduction in requirements for physical security measures during fabrication, transportation, storage, and use. All these reasons are connected with each other and cannot be considered individually.

The Reduced Enrichment Program of the United States

The U.S. Reduced Enrichement Research and Test Reactor (RERTR) Program includes six interacting technical elements. These are illustrated in Fig. 1-1 and described below.

1.3.4.1 Evaluation of HEU Export Requests

This activity provides the U.S. Executive Branch with a technical evaluation of every significant request for export of highly enriched uranium (HEU).

The technical and economic justification of need for HEU submitted with each Export License Application is reviewed by the Argonne National Laboratory (ANL) and a short lead-time technical evaluation is perfomed for the specific reactor(s) for which the application is made. Each evaluation addresses the potential of the reactor(s) for conversion to reduced-enrichment fuel and provides the Executive Branch with a technical analysis of the tradeoffs among experiment performance, core lifetime, economics and licensing issues.

1.3.4.2 Generic Reactor Analysis and Design

This activity provides generic core analysis and design (physics, safety, thermal-hydraulics, structures and fuels) and reactor-facility analysis and design (heat rejection, hydraulics) studies of the major types (U, U_3O_8, or UAl_x/H_2O, U-ZrH/H_2O, UO_2/H_2O, and U-Al/D_2O) of research and test reactors with reduced enrichment. Performance and fuel cycle cost implications, and the problems associated with plutonium production and fuel supply, are addressed. For each reactor type, in-depth redesign studies are undertaken for representative existing reactors to evaluate the potential for converting them from the use of highly-enriched uranium fuel to the use of reduced uranium enrichment. In-depth design studies are performed also for new research and test reactors in the design phase, to evaluate reduced-enrichment fuel alternatives. Collaborative studies with personnel from the reactor projects involved are carried out as appropriate.

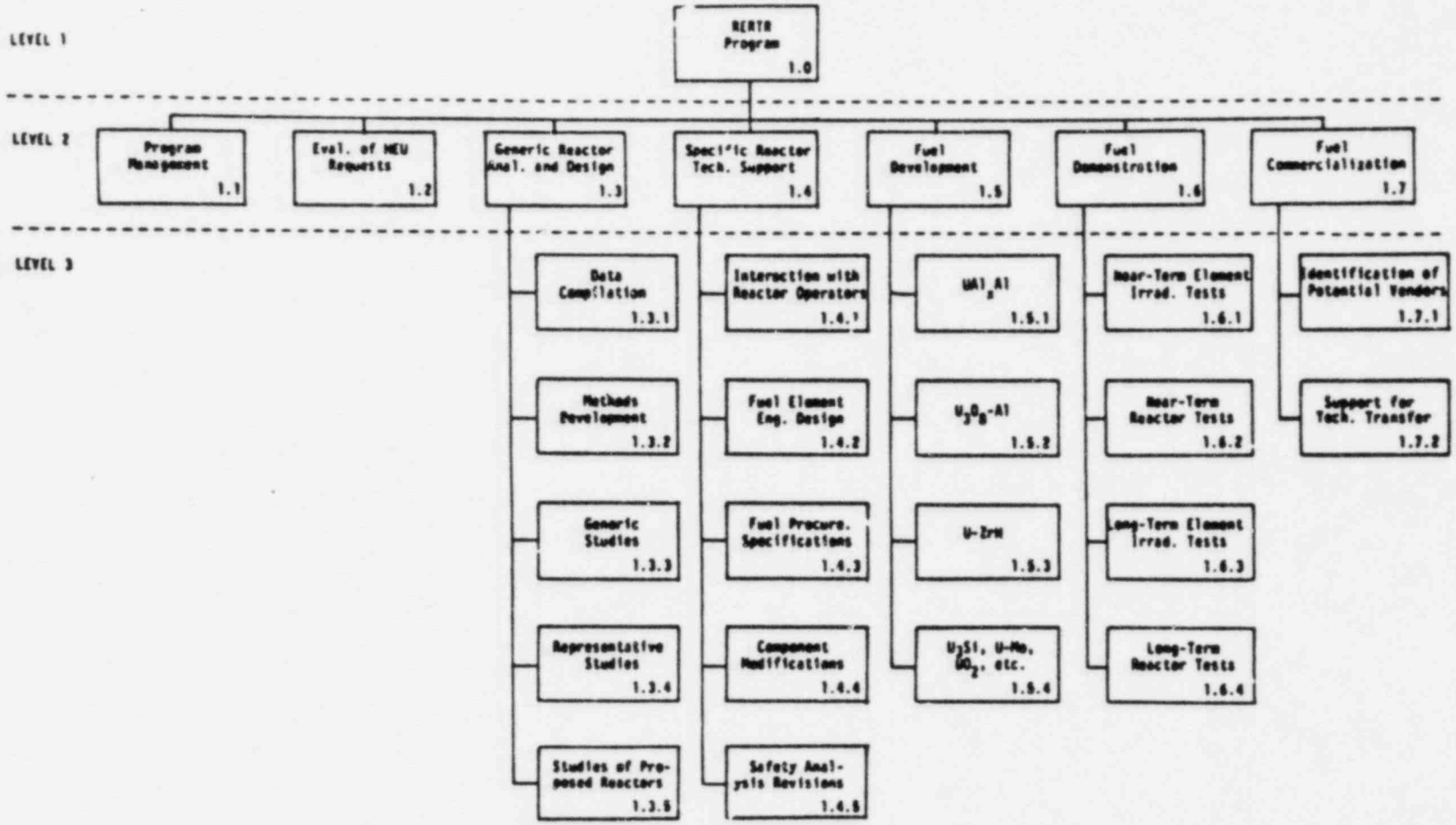
1.3.4.3 Specific Reactor Technical Support

This activity is structured to expedite application of reduced enrichment replacement fuel to specific foreign and domestic reactors by providing technical support to the fuel element engineering design, component design, procurement specification preparation, and safety analysis revisions necessary to initiate fuel procurement. Wherever possible, the support work is carried out in close cooperation with the affected reactor operating organization and fuel manufacturers. If appropriate and contributory to expediting priority applications, drawings and other documents supporting the procurement specifications may also be provided by ANL to the reactor operating organization. Technical support during procurement negotiations and fuel fabrication are provided by ANL, if necessary.

1.3.4.4 Fuel Development

This activity is a long-term fuel development effort intended to yield fabrication techniques for research and test reactor fuels of high uranium density. The fuel development activity consists of four parallel fuel development efforts. Three of these efforts are concerned with development of plate-type UAl_x-Al fuel elements, plate-type U_3O_8-Al fuel elements, and rod-type U-ZrH_x (TRIGA) fuel elements with uranium loadings much greater than those currently available. These three efforts are further developments of fuels that are now utilized in research and test reactors. The fourth effort is the development of new research and test reactor fuels (such as U_3Si, U-Mo, UO_2) that accommodate very high uranium loadings beyond the development potential of current fuels.

Figure 1-1.. Work Breakdown Structure of Reduced-Enrichment Research and Test Reactor (RERTR) Program



1.3.4.5 Fuel Demonstration

The objective of this activity is to demonstrate to the users and operators of research and test reactors that the operation of such reactors with reduced uranium enrichment fuels meets all the required criteria of reliability, performance, safety, core lifetime, and economics. The fuel demonstration activity includes three types of tests. The first test type consists in irradiating in a high-flux facility some elements of each relevant fuel type beyond their normal life burnup limit, and in verifying the ability of the fuel to stand such a test with acceptable metallurgical performance. The second test type consists of a whole-core demonstration in a reactor in which detailed physics measurements can be made to assess any change in the physics and safety characteristics of the core. The third test type consists of a whole-core demonstration in a reactor in which the burnup rate is sufficient to adequately study the physics/safety characteristics of the core throughout the entire fuel cycle. The fuel demonstration activity includes the planning of the tests, the procurement of the fuel elements/cores for the tests, the performance of the irradiations and experiments, post irradiation examinations, and analysis of data.

1.3.4.6 Fuel Commercialization

This activity is to provide the technical support to ensure that the fuel needed for the operation of all research and test reactors which can operate with reduced-enrichment fuel can become commercially available, on a worldwide basis, and without the need for significant government financial support. This part of the program includes: (1) identification of the potential commercial domestic and foreign suppliers of reduced-enrichment fuel for research and test reactors, (2) evaluation of their fabrication processes and capabilities, and (3) technical support and implementation for the transfer of technologies, wherever such transfer is appropriate and may contribute to the commercialization goal.

1.4 MAIN OPTIONS AVAILABLE FOR CONVERSION

1.4.1 General Technical Basis to Achieve Conversions Meeting Desired Criteria

In assessing the practical feasibility of utilizing lower enriched fuel in existing research reactors, the agreed criteria are that the safety margins and fuel reliability should not be lower than for the current design based on highly enriched uranium, major reactor modifications should not be required, and that preferably neither any loss in the overall reactor performance (e.g., flux-per-unit power) nor any increase in operation costs should be more than marginal. It is also recognized that the feasibility of reduced-enrichment use in each specific reactor must be objectively assessed on an individual basis taking into account all technical, programmatic, economic and licensing factors. However, it should be noted that there are specific applications requiring high flux reactor operation that can only be met with high enrichment fuel.

Enrichment reduction by simple substitution of lower enriched uranium in existing fuel designs has the immediate effect of reducing core performance and cannot meet the above criteria. Core reactivity is decreased, and therefore fuel burnup capability is decreased and fuel costs are increased, and/or core size is increased and therefore flux-per-unit power performance is decreased.

Enrichment reductions are feasible for most research and test reactor designs when the ^{235}U content in the fuel element can be kept approximately the same while the enrichment is decreased, or when it is increased, so that the reactivity loss due to the greater ^{238}U content is compensated to provide adequate lifetime.

Matching ^{235}U content (i.e., maintaining the same ^{235}U weight in each fuel element) would result in in-core flux-per-unit-power performance comparable to that of the unmodified reactor but, because of the poisoning effect of ^{238}U, would generally result in lower reactivity and reduced burnup potential. Burnup potential can be matched to that of the unmodified reactor by increasing the ^{235}U content in the reduced-enrichment core by some amount over that of the 93% enriched case at the expense of some decrease in in-core thermal-flux-per-unit-power performance. The importance of these flux effects is dependent on the particular reactor, the type of application, and conversion scheme adopted. For example, thermal flux decreases in the reflector and in flux traps are generally much less than in-core. Another possibility is to reduce costs by increasing the fuel cycle length. This could be accomplished by further increasing the ^{235}U content.

The increase of the overall uranium content per fuel element can be achieved by increasing the volume fraction of the fuel meat and/or by increasing the uranium concentration in the fuel meat.

Increasing the volume fraction of the fuel meat normally requires redesign of the fuel element. Three options are open: decreasing the clad thickness, decreasing the coolant volume fraction and/or decreasing the number of plates per element. The achievable reduction in the clad thickness may be limited by the minimum thickness needed for fission product retention. The achievable reduction in the coolant volume fraction may be limited by the need to avoid excessive pressure drop in the core and by the need to adequately moderate the neutron flux in the core. Otherwise the excess reactivity and cycle length would be significantly reduced. The reduction in the number of plates may be limited by the minimum heat transfer surface needed to prevent onset of nucleate boiling at a given reactor power.

These limitations may make it difficult to significantly increase the fuel meat volume fraction in some high-performance reactors that are designed very close to their thermal-hydraulic limit. In a majority of the research and test reactors in operation, however, and especially in those of low power, the volume fraction of the fuel meat can be increased above current values. Sometimes, a practicable way seems to consist in increasing the fuel meat thickness and coolant channel width by the same fraction, thereby reducing the number of fuel plates correspondingly. This is illustrated in Section 2 in more detail.

Increasing the uranium concentration in the fuel meat without changing the meat thickness has only negligible effects on the thermal-hydraulics properties of the core, and, therefore, it does not normally require redesign of the fuel element. (Only in some very rare cases might it be desirable to increase the coolant volume fraction to balance the hardening of the neutron spectrum caused by the increased uranium content). The only limitation to this approach is posed by the highest uranium concentration feasible with the most advanced fuel fabrication technology. This approach can be immediately applied to all those research and test reactors in which the uranium density in the fuel meat is less than currently qualified technology allows. Its application in reactors which already use the most advanced currently qualified fuel fabrication technology requires development of new fabrication techniques yielding even greater uranium densities in the fuel meat. Development of the new fabrication techniques is currently underway in the U.S. RERTR Program, in the French Reduced Enrichment Program, in the Reduced Enrichment Program of the Federal Republic of Germany, and also at the CNEA in Argentina, but it is anticipated that the desired fuel properties will be achieved only after several years.

For the rod-type $UZrH_x$ fuel, enrichment reduction is achieved by an increase in the uranium concentration in $UZrH_x$ alloy. The geometry of the fuel elements remain identical to the highly enriched version replaced.

The main properties of the currently qualified fuels and the status and development potential of the new fuels are summarized in the next section. More detailed information on the fuel development programs is provided in Section 3.

1.4.2 Status of Current, Near-Term, and Long-Term Fuel Technologies

Fuel meat materials currently qualified for use in research reactors are:

(1) U-Al Alloy, with uranium densities up to 1.1 g/cm^3.

(2) UAl_x-Al Dispersions, with uranium densities up to 1.7 g/cm^3.

(3) U_3O_8-Al Dispersions, with uranium densities up to 1.7 g/cm^3.

(4) U-ZrH_x, with uranium densities up to 1.3 g/cm^3.

Excellent burnup experience has been acquired on these fuels, albeit with uranium enrichment frequently greater than 20%. The enrichment is not expected to affect in any significant manner the metallurgical performance of the fuel, and tests already in progress are anticipated to prove conclusively that the experience gathered with these fuels does not depend on the fuel enrichment.

(5) UO_2 with density of 9.1 g U/cm^3 is currently used with rod cluster geometry. This fuel is qualified with plate-type geometry (Caramel) in low and medium power range and is under demonstration for high power reactors.

A high potential exists for increasing the maximum loading of many of these fuel types significantly above currently qualified values. In addition, greater uranium loadings can be achieved through the development of new fuel types, such as U_3Si and U-Mo. An overview of the development potential of the various fuel types is provided in Table 1-1, and the anticipated dates of commercial availability of suitably-qualified fuels are given in Table 1-2.

1.5 MAIN ACTIVITIES NEEDED IN PREPARATION FOR A TYPICAL CONVERSION

Several technical activities must be accomplished before a reactor conversion from the use of HEU fuel to the use of LEU fuel can be physically implemented. Because of their nature, a few of these activities are the exclusive responsibility of the organization to which the reactor to be converted belongs. Most of the activities may be shared, however, to a greater or lesser extent, with other organizations equipped with the needed expertise, resources, and willingness to assist in the conversion process. It is especially in this connection that the various national reduced enrichment programs can provide conversion assistance to the research and test reactor community, through IAEA coordination.

1.5.1 Characterization of Present Performance

Identification of key characteristics of reactor performance with the fuels currently utilized must be made. This, of course, must be responsibility of the reactor organization. This information is needed to identify any unique characteristics and special requirements of the reactor, and to establish a reference against which calculations with reduced enrichment may be compared. Needed information would include, for instance, the power distribution in the core, the neutron spectrum, the temperature coefficients of reactivity, the control rod worths, the thermal-hydraulic margins, the core lifetime, etc. Much of this information may be already available; however, collection and organization of the data in a form suitable for the intended purpose may be needed. In addition, experimental determinations may be needed in those cases in which the data are not available. Appendix G summarizes the reactor data normally needed as a basis for reduced enrichment conversion studies.

Table 1-2. Anticipated Dates of Commercial Availability of Suitably-Qualified REU Fuels

Fuel System	Uranium Density g/cm^3	Meat Thickness mm	Date of Availability
UAl_x-Al	2.6	0.5 - 1.5	1983
U_3O_8-Al	3.0	0.5 - 1.5	1983
	3.2 - 3.5	0.5 - 1.5	1985
UO_2-plates	9.1	$\geq$1.4	1980
	4.5	1.2	1983
UO_2-rods	9.1	8.2*	1980
U-ZrH_x	3.7	13.7*	1980
U_3Si-Al	4 - 8	0.5 - 0.8	1986

*Rod Diameter.

1.5.2 Performance Calculations with MEU and LEU

Before the conversion to reduced enrichment is studied in detail, the priority of design criteria for the conversion has to be specified. Possibilities include: minimum reactor core modification, minimum changes in operational characteristics and neutron flux values, minimum licensing problems, minimum fuel cycle costs, reoptimization for highest performance under certain boundary conditions (given maximum flow or power).

When the target is specified, the different options available should be compared which allow enrichment reduction to 20% or less. If no option satisfies the requirements, 45% enrichment would be considered.

This study will generally require calculations of the neutronic and thermal-hydraulic performance of the reactor with some parameter variations. The reactor data discussed in Section 1.5.1 must be calculated for the design variations considered to accompany the fuel enrichment change. Neutronics considerations include composition and thickness of the fuel meat, clad thickness, number of plates or rods per element, core size, fuel management strategy, etc.

THE STATUS AND DEVELOPMENT POTENTIAL OF RESEARCH AND TEST REACTOR FUELS

The concern about the proliferation potential of HEU fuels and about anticipated restrictions on HEU supplies has stimlated development programs on fuels with higher uranium content which would allow the use of uranium of lower enrichment. Fuel development programs are underway in the U.S., Canada, France, the Federal Republic of Germany, Japan, and Argentina.

The fundamental objective of these fuel development programs is to develop existing and new research and test reactor fuels of both plate-type and rod-type to their maximum feasible uranium loading, with the intent of improving the performance of reduced-enrichment reactors.

H.1 PLATE-TYPE FUELS

A variety of fuel element materials are under development for plate-type fuels. Some of these materials correspond to extensions of materials which are in current use, while others are entirely new. The enrichment reduction potential of the current and new fuels are shown in Table H1. It is evident from the table that extensions of currently utilized fuels will permit enrichment reductions to <20% enriched fuel in low and high power research and test reactors, but that only the new fuels will permit such reductions for very high power reactors. It is also evident from the table that enrichment reduction to <20% for low power reactors fuels could be accomplished with existing technology.

In the following sections, the presently utilized fuels and the new fuels are characterized by uranium content and performance, and the limits of uranium loading of fuels for plate-type reactors are estimated.

Table H1. Uranium Density and Enrichment Reduction Potential of Candidate Fuels for Research and Test Reactors with Plate-Type Fuels

Fuel Type	Current Uranium Loading, g/cm^3	Near-Term Uranium Loading, g/cm^3	Long-Term Uranium Loading, g/cm^3	Current/Near-Term/Long-Term Enrichment Reduction Potential, %		
				Low-Power Reactors	High-Power Reactors	Very High-Power Reactors
U-Al Alloy	1.1	1.3	~1.6	<20	70/45/45	93
UAl_x-Al	1.7	2.2-2.6	2.6-2.8	<20	45/20/20	93/45/45
U_3O_8-Al	1.7	2.2-3.3	3.3-3.8	<20	45/20/20	93/45/45
UO_2 Cermet	9.1[a]	-	-	<20	<20	<20[b]
U_3Si-Al	-	4.2-4.8	7.0-8.0	<20	93/20/20	93/45/20
U_3Si (bulk)	-	-	~11	<20	93/93/20	93/93/20

[a] 8.7 if the zircaloy spacers are smeared within the fuel meat. The density of the UO_2 is 10.3 g/cm^3.

[b] For very high-power reactors, UO_2 would have to be fabricated in very thin sections to provide proper heat removal.

prepared by ANL

INTRODUCTION

General Atomic Company has developed shrouded 4-rod and 16-rod clusters utilizing the TRIGA low-enriched uranium zirconium hydride (UZrH) fuel for use in converting and upgrading existing MTR plate-type reactors and also for fueling new TRIGA reactors. The use of low-enriched uranium is in keeping with non-proliferation policies and is readily exportable. The 4-rod cluster is designed to operate at power levels up to 3 MW and the 16-rod cluster is designed for power levels up to 10 MW in existing reactor core structures.

Both types of clusters use fuel-moderator rods which contain the well proven UZrH fuel in an Incoloy cladding. The rod diameter in the 4-rod cluster (3.24 cm) is only slightly smaller than that used in standard TRIGA fuel for more than 20 years. The 16-rod cluster uses a rod of 1.295 cm diameter and is identical in design to the fuel rods used in the 14 MW TRIGA now in operation at the Romanian Institute for Nuclear Technology. The fuel alloy used in the 4-rod cluster contains 20 wt-% uranium and in the 16-rod cluster 45 wt-% uranium. This provides a very high U-235 content with low enrichment, i.e., 440 grams U-235 in the 4-rod cluster and 880 grams U-235 in the 16-rod cluster. A small amount of erbium is included as a burnable poison and is a major contributor to the prompt negative temperature coefficient, the dominant safety feature of the TRIGA fuel. The high uranium loading combined with the burnable poison result in a very long burnup lifetime and favorable fuel cycle economics.

This Appendix is divided into two parts: B.1, which describes a 2 MW reactor using the 4-rod cluster and B.2, which describes a 10 MW reactor using the 16-rod cluster.

4.5 PROMPT NEGATIVE TEMPERATURE COEFFICIENT

The basic parameter which provides the great degree of safety in the operation of a TRIGA reactor system is the prompt negative temperature coefficient. This temperature coefficient (α) allows great freedom in steady-state operation, since the effect of accidental reactivity changes occurring from experimental devices in the core is minimized.

The prompt negative temperature coefficient for the TRIGA-LEU core is based on the same core spectrum hardening characteristic that occurs in a standard* TRIGA core. The spectrum hardening is caused by heating of the fuel-moderator elements. The rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, the thermal neutron spectrum in the fuel element shifts to a higher average energy (the spectrum is hardened), and the mean free path for neutrons in the element is increased appreciably. For a standard TRIGA element, the average chord length is comparable to a mean free path, and the probability of escape from the element before being captured is significantly increased as the fuel temperature is raised. In the water the neutrons are rapidly re-thermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel in a standard TRIGA element thus causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature-dependent disadvantage factor for the unit cell in which the ratio of absorptions in the fuel to total cell absorptions decreases as fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity.

In the 4-rod cluster TRIGA-LEU fuel, the temperature-hardened spectrum is used to decrease reactivity through its interaction with a low-energy resonance material. Thus, erbium, with its double resonance at $\sim$0.5 eV, is used in the TRIGA-LEU fuel both as a burnable poision and as a material to enhance the prompt negative temperature coefficient. The ratio of the absorption probability to the neutron leakage probability is increased for the 4-rod cluster TRIGA-LEU fuel relative to the standard TRIGA fuel because the U-235 density in the fuel rod is about 2.5 times greater and also because of the use of erbium. When the fuel-moderator material is heated, the neutron spectrum is hardened, and the neutrons have an increasing probability of being captured by the low-energy resonances in erbium. This increased parasitic absorption with temperature causes the reactivity to decrease as the fuel temperature increases. The neutron spectrum shift, pushing more of the thermal neutrons into the Er-167 resonance as the fuel temperature increases, is illustrated in Fig. 3 where cold and hot neutron spectra are plotted along with the energy dependent absorption cross section for ER-167. As with a standard TRIGA core, the temperature coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator; thus, fuel and solid moderator temperatures rise sumultaneously, producing the temperature-dependent spectrum shift.

*A standard TRIGA core contains U-ZrH fuel with no erbium. The uranium enrichment is 20%, and the fuel element (rod) diameter is about 3.8 cm (1.5 in.) with a core water volume fraction of about 0.33.