DECOMMISSIONING PLAN

THE CATHOLIC UNIVERSITY OF AMERICA

AGN-201 REACTOR Serial No. 101

License No. R-31
Docket No. 50-77

DECEMBER 1991

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DECOMMISSIONING PLAN

for the

CATHOLIC UNIVERSITY OF AMERICA AGN-201 NUCLEAR REACTOR

1.0 ORIENTATION

1.1 Introduction

In accordance with 10 CFR 50.82, this dismantling and decommissioning plan is submitted to the U.S. Nuclear Regulatory Commission (NRC)[1] in support of the request of The Catholic University of America (CUA) for authority to surrender License R-31 voluntarily and to decommission the CUA AGN-201 Nuclear Reactor (CAGN) and dispose of its components. The plan describes the means by which all radioactive or contaminated components will be removed and the facility will be decontaminated. The plan provides reasonable assurance that dismantling the facility and disposal of its components will be performed in accordance with the Code of Federal Regulations (CFR) and will not be inimical to the common defense and security nor to the health and safety of the public.

The format of this plan has been adapted from that which previously had been accepted with respect to the University of Utah AGN-201M reactor. The plan includes information on the history of the facility, its current radiological status, dismantling alternatives, the decommissioning organization and its responsibilities, the regulations, regulatory guides and standards which will guide the decommissioning activities, and the qualifications of the decommissioning staff. The plan describes the personnel protection program, tasks and schedules, physical security and safeguards, radioactive materials management, technical specifications, environmental impact and the termination radiation survey plan which together comprise the decommissioning plan.

In anticipation of decommissioning, the fuel has been removed from the reactor and is stored in the CUA fuel storage facility, pending transfer to its owner, the Department of Energy (DOE). DOE has stated tentatively that the fuel will be shipped to the DOE Operations Office (ORO), Y-12 Facility, Oak Ridge, TN. The fuel will be shipped under the existing license and in accordance with DOE, NRC, and Department of Transportation (DOT) requirements as soon as final DOE instructions for transfer are received. CUA has received a scrap declaration form and related forms from ORO. It is expected that shipping containers and consignment instructions for the fuel will be provided by ORO, in a timely manner, following completion

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[1] A list of all abbreviations and acronyms used is included as Appendix J.

and return of the forms by CUA. As per 10 CFR 50.59 assessment by CUA, shipment of the fuel represents no unreviewed safety questions and the fuel will be shipped under the current license as soon as practicable.

The decommissioning effort will be managed by the CUA Radiation Safety Office (CRSO). The Director of Radiation Safety (DRS) has been designated Reactor Administrator (RA) for the purpose of decommissioning. The staff consists of the Director, who is the CUA Radiation Safety Officer (RSO) and one full-time health physics technician (in training to become a health physicist); two part-time personnel provide electronics maintenance, and computer entry and clerical assistance. It is anticipated that all CUA decommissioning activities can and will be performed by these University personnel. This decommissioning plan and all prior decommissioning activities have been reviewed and approved by the CUA Radiation Safety Committee (RSC).

1.2 Synopsis of the Decommissioning Plan

The selected decommissioning method is D E C O N through which the reactor structure will be dismantled--to the extent necessary to insure that the radioactivity of its components is consistent with their release for unrestricted use or disposition, or to identify components which must be disposed of as radwaste. No other decommissioning alternative is considered to be reasonable. The cost of the decommissioning project is expected to total about $11,000, including the cost of the termination radiation survey, and not including disposition of the decontaminated reactor structure. Post-decommissioning disposition of the structure is the responsibility of the Mechanical Engineering Department. It is anticipated that it will be cut up for scrap, in pieces whose size will permit either burial in the present reactor pit or removal from the building via existing doorways. Decommissioning funding has been committed by the CUA Administration via the normal budget of the CRSO. This decommissioning plan applies to the CAGN, its structure and components, and to Room B-16R, Pangborn Hall, including radioactive waste that may be generated during the decommissioning process. The criteria for unrestricted release shall be the limits recommended in NRC Regulatory Guide (RG) 1.86. In addition, the exposure rate due to surface contamination shall be less than 5 uR/h above natural background at a distance of one meter. Any components which cannot be decontaminated to comply with these criteria will be transferred to the CRSO for disposal in accordance with applicable federal, state, and local regulations. Decommissioning activities will be performed under the existing CUA radiation safety program to ensure that occupational exposures are ALARA. These activities also will be performed in conformance with established industrial safety practices. All activities will be completed within one year of receiving approval from NRC to proceed with decommissioning. No special equipment, tools, or contractor assistance will be required and quality assurance reviews and audits will be conducted as needed by the RSC. All solid and liquid radioactive wastes generated during the project will be transferred to the CRSO for disposal. The collective dose equivalent to decommissioning staff is anticipated to be less than 0.1 person-rem. A

final report will be submitted to NRC, requesting termination of the CAGN possession-only license certifying completion of a comprehensive termination radiation survey which will verify that the facility, including residual CAGN components, satisfies the criteria for release to unrestricted use.

1.3 Facility History

The original application to acquire and possess an AGN-201 nuclear reactor was filed by CUA on 10 July 1957 and amended on 23 August and 13 November 1957. Pursuant to License No. R-31, issued 15 November 1957, AGN-201, Serial No. 101, was acquired from the Aerojet-General Nucleonics Corporation (AGN). The original license was for a period of twenty years and specified a maximum thermal power of 100 milliwatts. Initial criticality was achieved on 20 November 1957. The license has been amended nine times. The more significant amendments were Number 3, dated 26 April 1961, which allowed relocation of the reactor to the newly constructed Pangborn Engineering Building from its original location in the Power Plant Building, Number 7, of 14 March 1979, which extended the operating license to the year 1997, and Number 9, of 5 March 1986, which changed its status from operational to possession-only. A description of the CAGN is provided in Appendix A.

The CAGN is located in the Pangborn Engineering Building in the northeast part of the campus (Fig. 1). The Reactor Room (B16-R), is located on the basement floor of Pangborn Hall (Fig. 2) and comprises approximately 1000 square feet of the northeast corner of the basement floor of the north wing of the building. The cinder block wall which had been proposed to divide B16R, creating a new room B16A (as shown in the figure), was never built. Therefore the reactor facility consists of the spaces shown in the figure as B16A and B16R. Room B18-A (Fig. 2) houses the CRSO and a portion of the Radiation Safety Laboratory. These are the only two restricted areas in Pangborn Hall. Both areas are under the control of the DRS.

The reactor was operated first by the Department of Nuclear Science and Engineering, and subsequently by the Department of Mechanical Engineering, for training nuclear engineering graduate students. The reactor usually was operated in conjunction with a Nuclear Reactor Laboratory course and used primarily for performing basic reactor physics experiments. The reactor was operated at power levels up to one hundred milliwatts.

The total thermal power produced during operations under License No. R-31 is estimated to have been 16.2 watt-hours. (It is understood that the reactor had been operated at various times and places prior to its receipt by CUA. No record of these operations is held.) Appendix B provides an annual summary of operating time and power at CUA.

The reactor was last shut down in December of 1982. Surveillance required by the Technical Specifications has been continued. The reactor was defueled on 17 March 1990, in accordance with a detailed defueling plan which had been approved by the RSC and

submitted to the NRC. Appendix C contains an estimate of the fission product inventory of the fuel resulting from operating the CAGN at CUA.

1.4 Radiological Status of the Facility

The CAGN had a polyethylene-moderated, graphite-reflected, water-shielded core. The reactor core consisted of a set of nine circular polyethylene disks, approximately 25 centimeters in diameter, containing a uniform dispersion of uranium oxide enriched to less than 20% U-235. Both safety rods and the coarse control rod contained similar fuel material, in the form of columns of right circular cylindrical segments, collectively sealed in aluminum capsules, so that reactivity increased as rods were inserted into the reactor core; the fine control rod consisted of aluminum-encapsulated polyethylene (unfueled). Monthly radiation surveys prior and subsequent to defueling have shown no evidence of radioactive contamination due to activation of reactor components or the leakage of fission products.

The 10 mCi radium-beryllium start-up neutron source was removed and placed in a neutron source storage tank prior to removal of the fuel from the reactor, to keep radiation exposures ALARA during defueling. Wipe tests of the sealed source gave no indication of leakage. The source was shipped to RAMP, Inc. in Denver, CO, in the summer of 1990 for disposal at the low level waste site at Beatty, NV.

Following removal of the start-up source, defueling proceeded as per written procedure. Wipes of reactor components adjacent to the core can and of the exterior of the core can showed no evidence of contamination or fission-product leakage. The core can was removed from the reactor and transferred to the fume hood in the Radiation Safety Laboratory. A gaseous sample was taken from the hermetically-sealed core can and analyzed for the presence of radioactive contamination. No fission products were observed in the gamma spectrum of the sample. The control rod guide tubes were removed from the core can and replaced with pre-fabricated cover plates to permit insertion of the core can into a storage safe. At a later date, the fuel disks were removed from the core can, inspected, surveyed, inventoried, sealed in plastic bags, and re-stowed on two separate shelves of the safe. Calculations performed by the RA confirmed that the value of k_{eff} for the storage array in the fuel safe was much less than the value of 0.8 required by the Technical Specifications. Fuel from the safety and control rods was similarly inspected, surveyed, inventoried, sealed, and stored in a separate safe in the Reactor Room. Radiation levels around the reactor structure decreased to nominal background levels promptly upon removal of the core can.

Upon removal of the core-can from the reactor, the fuel had a maximum container surface exposure rate of 4.5 mR/h and a minimum of 0.1 mR/h. The low exposure rates are indicative of the low burn-up of the fuel, the low neutron flux in the reactor at full licensed power, and the fact that the reactor had not been operated since 1982 (approximately seven years). Because both the neutron

flux and usage were low, no measurable contamination and/or acti-ation has been found during extensive surveys performed by the CRSO. A minor amount of fixed and removable contamination is ex-ected to be found on surfaces of reactor core components which were in direct contact with the fuel. All such components have been sealed in plastic, and remain stored in the reactor room, pending furthe. survey. These components will be disposed of in accordance with RG 1.86 after NRC approval to decommission has been received.

1.5 Decommissioning Alternatives

The Catholic University of America is seeking authority to surrender License R-31 and revert the facility to unrestricted use. The University has selected D E C O N as the method for decommissioning the CAGN. It is expected that all reactor equipment and structures containing radioactive contaminants will be decontaminated to levels which permit their release for unrestricted use. It is the intention of the licensee to remove all radioactive fluids, radioactive waste, and other materials connected with the mechanism, operation, and decontamination of the CAGN and reassign the reactor room for unrestricted use. The reactor pit and the reactor room floors are vinyl-tile-surfaced concrete, slab-on-grade. There are no sewer drains in the facility. No other spaces are pertinent to this decommissioning plan. Any materials or components which do not satisfy the conditions for unrestricted release will be transferred to the CRSO for disposal as radioactive waste. Although not part of the decommissioning plan, it is intended to cut the decontaminated structure into pieces small enough for either removal through existing doorways or burial in the reactor pit, which would be back-filled and topped by a concrete slab, continuous with the present reactor room floor.

No other decommissioning alternative is reasonable. The reactor has not been used in over eight years; it is of no present (or perceived future) value to CUA. It is desired to use the space occupied by the CAGN for other purposes. There is no known market for the intact reactor structure and/or control console. To remove the structure, intact, would require removal and replacement of a portion of the exterior wall of the reactor room, a costly effort. The Smithsonian Institution had expressed an interest in the structure and control console; they subsequently dropped the matter.

1.6 Organization and Responsibilities

The organization responsible for the decommissioning is diagrammed in Figure 3. Individuals at various management levels have responsibilities related to safeguarding the public and facility personnel from undue radiation exposure and for adhering to all requirements of the facility license. The assignment of specific responsibilities is described below.

President - The President is the Chief Administrative Officer responsible for the University and its activities.

Executive Vice President (EVP) - The EVP is responsible to the President for the day-to-day operation of the university and repre-

sents the President in all health and safety matters, including those pertaining to the reactor facility. The EVP promulgates the University Radiation Safety Program, with the advice of the RSC.

Reactor Administrator (RA) - The RA (who is also the DRS) is the University Administrative Officer responsible to the EVP for the reactor facility and for related Technical Specification compliance, safety and decommissioning, and for insuring regulatory compliance with respect to the reactor facility. In this capacity, and in conformance with the facility license and the policies set forth by the RSC, the RA prepares all regulations for the facility, reviews all procedures, seeks approval of all procedures and proposals for changes and experiments from the RSC, and is responsible for the health and safety of all personnel in the facility. The RA will prepare the final decommissioning report to the NRC.

Radiation Safety Committee RSC) - The RSC advises the EVP in all matters concerning the radiological health and safety of personnel who might be exposed to radiation produced by university-owned and/or -operated sources or equipment. The committee reviews and approves the University's Radiation Safety Program, for promulgation by the EVP. The committee is informed of any occurrences related to radiation health and safety and/or reactor safety which are reportable to any authorities outside the University, advises the EVP of such occurrences and makes recommendations to the EVP with regard to any such matters. The committee holds periodic meetings and has the responsibility and authority to conduct audits and reviews of the records of the RA and the DRS.

The decommissioning plan has been reviewed and approved by the RSC. The RSC has audited the records of the Reactor Administrator and RSC which relate to pre-decommissioning activities conducted to date, and will continue to audit related activities until the decommissioning is completed.

The current membership of the RSC is shown in Appendix D.

Director of Radiation Safety (DRS) - The DRS is the University RSO, who administers the University's radiation safety program and provides technical assistance to the RSC (the RSO serves *ex officio* as a member of the committee). The RSO is authorized and directed to promulgate and enforce such procedures as are necessary to assure compliance with applicable federal regulations and to ensure the accurate interpretation and effective implementation of policies established by the RSC. The DRS is the head of the CRSO and oversees the activities of its personnel.

Decommissioning Staff - The decommissioning staff consists of the RA and the Health Physics Technician assigned to the CRSO. The staff is responsible for performing all decommissioning activities. Contractor assistance will not be required. These personnel are:

Dr. Warren E. Keene, Reactor Administrator and Director of Radiation Safety,

Mr. Mohammad S. Saba, Health Physics Technician and Health Physicist in Training.

1.7 Staff Qualifications

All decommissioning-related activities have been and will be performed by the RA/RSO, assisted by the Health Physics Technician in the CRSO. The RA is well-trained in the use of radiological equipment and in pertinent decontamination work. The Health Physics Technician is in-training and will be appropriately supervised. The qualifications and training of the staff, relevant to the decommissioning effort, are summarized in resumes included in Appendix E.

1.8 Regulations, Regulatory Guides and Standards

Decommissioning-related operations will be governed by pertinent portions of relevant federal, state, and local regulations, regulatory guides, and standards associated with nuclear research reactor dismantling, safety, radiological and environmental health, and industrial hygiene. A list of such which are held and believed applicable is included as Appendix F. The list is not intended to be exhaustive; it is deemed adequate.

1.9 Cost and Funding

The Catholic University of America has committed to provide adequate financial support to ensure the safe and timely decommissioning of the CAGN. It is estimated that the total cost of decommissioning will be less than $12,000, due to the fact that contamination or activation of the reactor structure in excess of 10 CFR 20 and RG 1.86 limits has neither been found, nor is anticipated. This amount includes the anticipated cost of radwaste disposal and the termination radiation survey. Funding is via the budget of the Radiation Safety Office.

2.0 Tasks and Schedules

The CAGN decommissioning plan includes all tasks whose completion is required in order to remove all radioactive components from the CAGN facility so that any components remaining in the facility at the time of the final survey will be within the contamination limits of RG 1.86 and below 5 uR/h above background at a distance of one meter. A preliminary survey has indicated this is already the case.

TASK	STATUS/SCHEDULE
1. Sample & dispose of shield water	Completed
2. Remove startup source	Completed
3. Dispose of startup source	Completed
4. Remove core can	Completed

5.	Disassemble and inspect fuel segments	Completed
6.	Submit fuel scrap declaration and related forms to ORO	1 FEB 92
7.	Ship fuel to DOE designee	60 days after receipt of containers and shipping authority
8.	Disassemble reactor structure to the extent necessary to determine the radioactivity of all components	Decommissioning approval plus 6 months
9.	Decontaminate reactor components	Decommissioning approval plus 6 months
10.	Transfer all radwaste to CRSO for disposition	Decommissioning approval plus 6 months
11.	Termination radiation survey of the CAGN facility.	Decommissioning approval plus nine months
12.	Submit final report to the NRC requesting termination of the facility license.	Decommissioning approval plus 12 months

3.0 Personnel Protection

3.1 Hazard Analysis

There are no special health or safety considerations which are pertinent to the dismantling or decommissioning of the CAGN. Normal radiological health protection considerations are appropriate to fuel handling and component decontamination activity and will be observed. Except for a few components which were in direct contact with the fuel, no radioactive contamination or activation of the structure has been found in extensive preliminary surveys; therefore, radiation exposure of personnel will be minimal.

Removal of the start-up source and the fuel constituted tasks in which exposure of personnel to radiation was unavoidable. However neither whole body nor finger dosimeters recorded measurable exposure incident to these operations. Additional handling of the fuel will be required incident to quarterly inspection and inventory, measurements required for the ORO scrap declaration, and packaging the fuel for shipment. However it is unlikely that significant radiation exposures to personnel will result from any of these activities. No other decommissioning procedures will involve any radiation exposure above natural background. Involved personnel are provided film (beta-gamma-neutron) and extremity (beta-gamma) badges by the RSO, incident to their other duties; therefore monitoring of exposure incident to fuel handling will be included in their exposure records. No radiation exposures in excess of nominal background levels have been reported by the dosimetry vendor for

periods during which start-up source or fuel have been handled. It is estimated that a collective dose-equivalent of less than 0.1 person-rem (whole-body and/or extremity) will have been received by decommissioning personnel when decommissioning is completed.

3.2 Radiation Protection

The radiation protection program will be consistent with applicable portions of the publications listed in Appendix F. A copy of the University's Radiation Safety Manual is included as Appendix G. Responsibility for proper control of radiation hazards at the CAGN rests with the RSO.

All decommissioning activities will be conducted to comply with the ALARA principle to minimize radiation exposure of decommissioning staff and will be conducted within the scope of the University's existing radiation safety program. Radiation safety during all operations will be assured through personnel monitoring, surveys, and procedures that are reviewed and approved by the RSO.

The CRSO possesses, uses, maintains and calibrates various radiation detection and survey equipment. It includes 3 inch by 3 inch sodium iodide and Ge(Li) detectors connected to a multichannel analyzer for gamma spectroscopy, a windowless gas-flow proportional counter for alpha and beta particle detection, and portable survey meters. The portable survey meters, which are provided by the CRSO to all laboratories which require them, include Johnson Associates Model GSM-15, Ludlum Model ESP, and Eberline Model E-120 Geiger counter survey meters. A liquid scintillation detector is available in the CRSO.

No significant personnel exposures are anticipated since little contamination or activation has been found in preliminary surveys. Personnel exposure will be monitored with portable radiation detectors, individual film (whole body), and TLD (extremity) dosimeters, when the latter are appropriate.

3.3 Industrial Safety and Hygiene

Decommissioning the CAGN will proceed in a manner consistent with applicable OSHA and District of Columbia industrial safety requirements. The University Department of Environmental Safety (DES) maintains a continuing accident prevention program, monitors the campus for hazards to environmental health, and provides laboratories and individuals with advice and technical assistance for developing effective health and safety practices. Its accomplishments have been recognized by being named a 1991 recipient of an Award of Merit by the National Safety Council. DES conducts regular inspections of University facilities. The staff includes the Director (a Certified Industrial Hygienist), an Occupational Safety & Health Specialist, and a Chemical Inventory and Waste Manager. The Director is responsible to the EVP for promoting good health and safety practices within the University community. The Director is a member of the RSC.

Decommissioning personnel are obliged to adhere to health and safety practices outlined in the University's Chemical Materials Safety Manual. The manual contains general procedures for emergency response, laboratory safety, industrial safety, general safety, disposal of non-radioactive hazardous waste, and fire safety. It also incorporates the OSHA-mandated Chemical Hygiene Plan. Decommissioning personnel are familiar with the contents of this manual. A copy of the manual is included as Appendix H.

The most probable type of accident related to decommissioning is that which might be called "mechanical" and might result from either human error or mechanical failure. The probability of human error will be minimized by making adequate preparation for the work and by following predetermined procedures which will be discussed by participating personnel before the actual work is commenced. The probability of mechanical failure will be minimized by thorough inspection of all equipment in advance of its use.

Decommissioning staff are supplied with coveralls, gloves, safety glasses and safety shoes, to be used when appropriate; dust masks and respirators are available and their use is required when appropriate. Chemical solvents may be used to remove radioactive contamination from some reactor components. Appropriate ventilation will be provided. The use and disposal of solvents, which are not radioactively contaminated will be in accordance with standard University practice. Disposal of radioactively contaminated solvents will be in accordance with the policies of the CRSO.

4.0 SECURITY AND SAFEGUARDS

Two categories of storage safety are discussed, that for the fuel and that for residual radwaste pending its disposition off-site.

4.1 Fuel

All CAGN SNM (reactor fuel) held under license No. R-31, including the thermal fuse and the unirradiated fission plate, has been stored in locked fuel storage safes in the reactor room since it was removed from the reactor on 17 March 1990. It will remain there until it is packaged for shipment to DOE's designee. The reactor room is a controlled access area, equipped with a security system which has been described in prior communication with NRC.

Pending its transfer, the fuel is inspected and inventoried periodically, in accordance with the technical specifications, to ensure that the material is secure, has not been tampered with, and that the physical integrity of the fuel and its containers has not deteriorated.

After receiving appropriate shipping containers and shipment authority, CUA will transfer the fuel in accordance with pertinent safeguards requirements of 10 CFR 73.67(g) and DOT regulations.

4.2 Radioactive Waste

Reactor components removed during the defueling process have been surveyed and sealed in plastic wrapping and are stored in the reactor room, pending NRC approval to proceed with decommissioning.

After receipt of decommissioning authority, reactor-related contaminated material, parts or components which can not be decontaminated to RG 1.86 standards will be transferred to the CRSO, for interim storage and/or disposal--depending on the availability of low level waste site access.

The CUA byproduct material, special nuclear material and source material licenses are adequate to provide for lawful retention of residual radwaste until such time as it can be deposited in a low level waste site.

The CRSO radwaste handling and storage facility is described in Section 6.3.

5.0 Risk Analysis

Because all fuel has been removed from the reactor and is stored in a configuration which insures that $k_{eff} < 0.8$, the Ra-Be neutron source has been disposed of, and very little contamination and no activated materials have been found, no credible radiological accident can be postulated. Criticality can be achieved only by reassembling the core and control rod material in the presence of an adequate reflector-moderator. Such an occurrence by accident is impossible.

6.0 Radioactive Material Management

6.1 Fuel Disposal

All fuel has been removed from the CAGN and is stored as described in Sec. 4.1. The fuel will be shipped to DOE's designee as soon as appropriate approval for fuel transfer is received. Scheduling was discussed in Sec. 2.0.

6.2 Radioactive Waste Processing

No gaseous or airborne radioactive wastes have been observed in the CAGN facility or are expected to be generated during the decommissioning process. A small amount of aqueous radioactive waste (estimated to be less than three gallons) may be generated in the decontamination of reactor components. A much smaller amount of radioactively contaminated chemical solvents may also result from decontamination efforts. Liquid wastes will be rendered into a form that facilitates disposal, as directed by the RSO. Less than one cubic foot of solid contaminated waste (comprised of wipes, disposable gloves and protective clothing, and drop cloths) was generated during the defueling; it has been disposed of by the CRSO. It is anticipated that a total of less than 5 cubic feet of additional solid waste will be produced incident to decommissioning. The waste

will be disposed of through the University's low-level radioactive waste program managed by the CRSO, and described in Appendix G.

Except for core components which were in direct contact with the fuel, preliminary surveys have found no evidence of radioactive contamination in the facility. Reactor components not in direct contact with the fuel are not expected to be contaminated or show any evidence of activation. However, any parts, components, or material which do not meet the unrestricted release criteria described in Sec. 1.2 and 2.0 will be delivered to the CRSO--for disposal in a low level radioactive waste site, in accordance with NRC and DOT regulations, at a time when site access exists.

Mixed waste generation is not anticipated. However the University's Vitreous State Laboratory has extensive experience in radioactive and mixed waste management research. Separation of any small quantity of mixed waste, which might be generated, into its respective toxic and radioactive components is expected to be feasible.

6.3 Radioactive Waste Storage

The CUA radioactive waste handling and storage facility consists of a quonset hut of approximately 600 square feet, with a central ceiling height of twelve feet. It is not shown on the campus map (Fig. 1); it is located adjacent to the south and east wings of St. John's Hall, which is shown. It is equipped with a fume hood with HEPA-filtered exhaust and a stainless steel sink where all licensee sewer disposal of radioactive material originates. A 30,000 pound-force in-drum compactor is on order. It will be installed during the early part of 1992. Given the present radwaste generation rate and anticipated increases therein, it is estimated that CUA has the capacity to store radwaste on site for more than five years, should lack of access to a low level waste site make that necessary. Plans exist to enhance the security of the facility, consistent with the potential increase in the average radwaste inventory which will result from prolonged site unavailability.

7.0 TECHNICAL SPECIFICATIONS

The reactor has been defueled and the reactor license has been amended to "possession only," with appropriately revised technical specifications. No further changes in the technical specifications are believed to be necessary during the decommissioning project. (Certain of the Technical Specifications will become inapplicable when the fuel has been shipped.) Radiation monitoring in the CAGN facility is performed monthly as part of routine campus surveillance by the CRSO. Pending transfer of fuel off site, the CAGN fuel will remain secured in locked-safe fuel storage facilities in the controlled access reactor room and maintained in an array such that k_{eff} is less than 0.8 for all conditions of moderation and reflection. While onsite, the fuel will continue to be inspected periodically to ensure that the material is secure and that the integrity of the fuel and its containers is maintained. A copy of the current Technical Specifications is included as Appendix I.

8.0 ENVIRONMENTAL IMPACT

Experience gained during the decommissioning of similar AGN reactor facilities (e.g., Oregon State University [NRC License R-51], Memphis State University [R-127]], and the University of Oklahoma [R-53]), has demonstrated that such activities may be completed successfully with only negligible changes in the amounts of effluents that may be released offsite and without any significant increase in individual or cumulative occupational radiation exposure. Because decommissioning of the CAGN facility involves no significant hazard to the public or the environment no environmental impact statement has been prepared. Identification of the use of Room B-16R, Pangborn Hall as a reactor facility will be maintained in the permanent records of the CRSO.

9.0 Termination Radiation Survey

The termination facility survey will include the reactor room and all components, material and equipment associated with the CAGN facility which are to be released for unrestricted use. On completion of decommissioning, only normal background radioactivity will remain in the reactor room.

In the reactor facility, all surface areas larger than one square meter will be marked off in a grid of one meter square blocks. The survey will include a wipe sample of 100 cm^2 (nominal) taken within each block to identify removable contamination. Wipes for each grid location will be counted for alpha and beta contamination. Limits for removable contamination will be 200 dpm/100 cm^2 (beta/gamma) and 20 dpm/100 cm^2 (alpha) (RG 1.86). A beta-gamma survey also will be performed centered on each block at a distance of one meter from the surface. Residual radioactivity will be considered acceptable if measurements so-made are not more than 5 uR/h above background, where the reference background value is obtained as the highest of corresponding measurements of similar structural material made elsewhere in Pangborn Hall. Results of the survey and the procedures used to obtain and analyze data will be audited by the RSO and the RSC to ensure the accuracy and completeness of the data.

Small objects, for which a one-meter grid is inapplicable will be wipe tested and surveyed in a manner which is consistent with the procedure used by the CRSO for inspection of incoming packages of radioactive material. Limits for removable contamination and radioactivity will be those described in the preceding paragraph.

The CUA RSO maintains various radiation detection and survey instruments which may be used for the termination survey. This equipment was described in Section 3.2. Additional instrumentation may be acquired for the purpose. Instruments used for these surveys will be currently-calibrated in accordance with approved CUA RSO procedures, which are consistent with appropriate ANSI Standards.

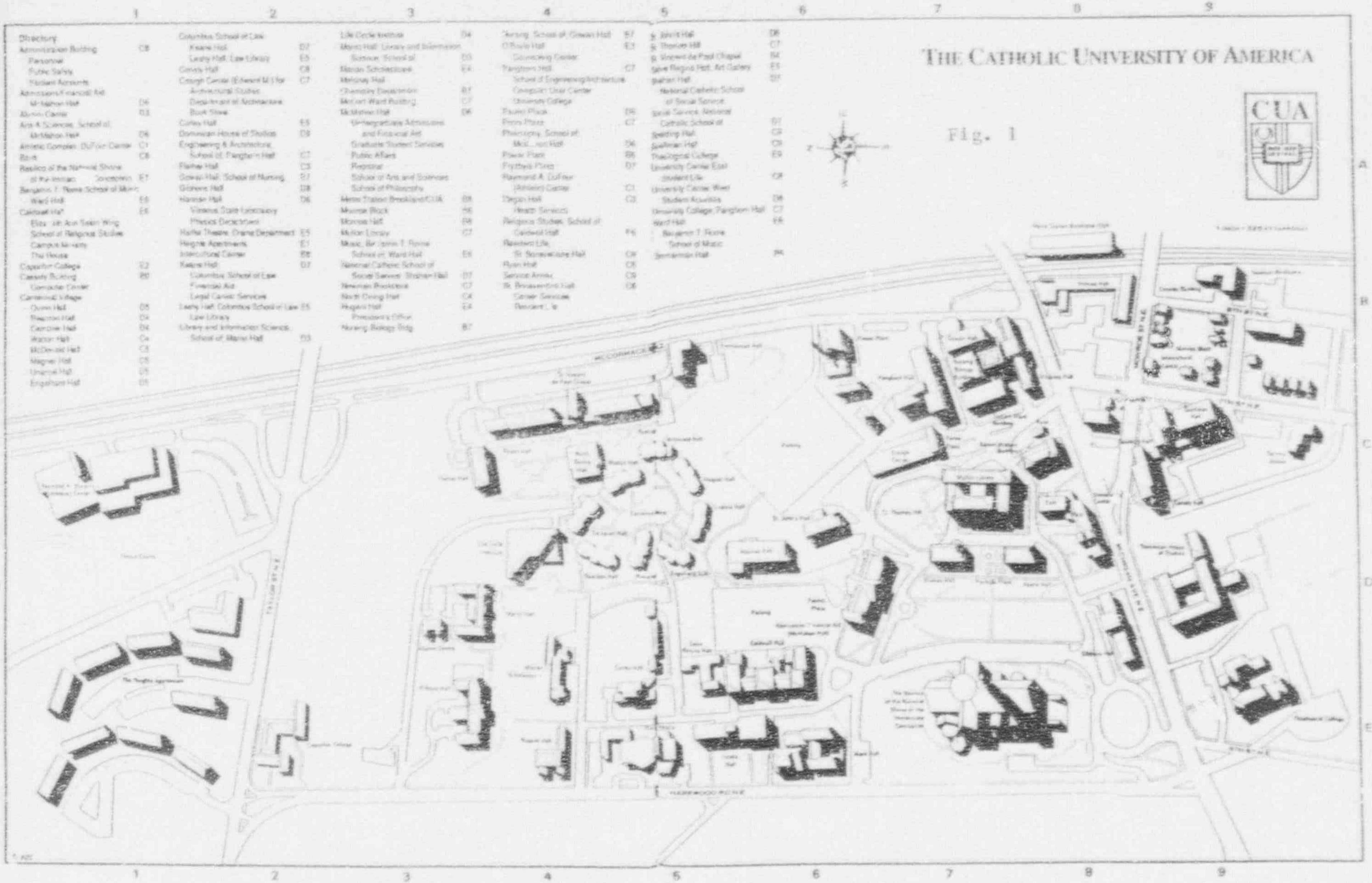
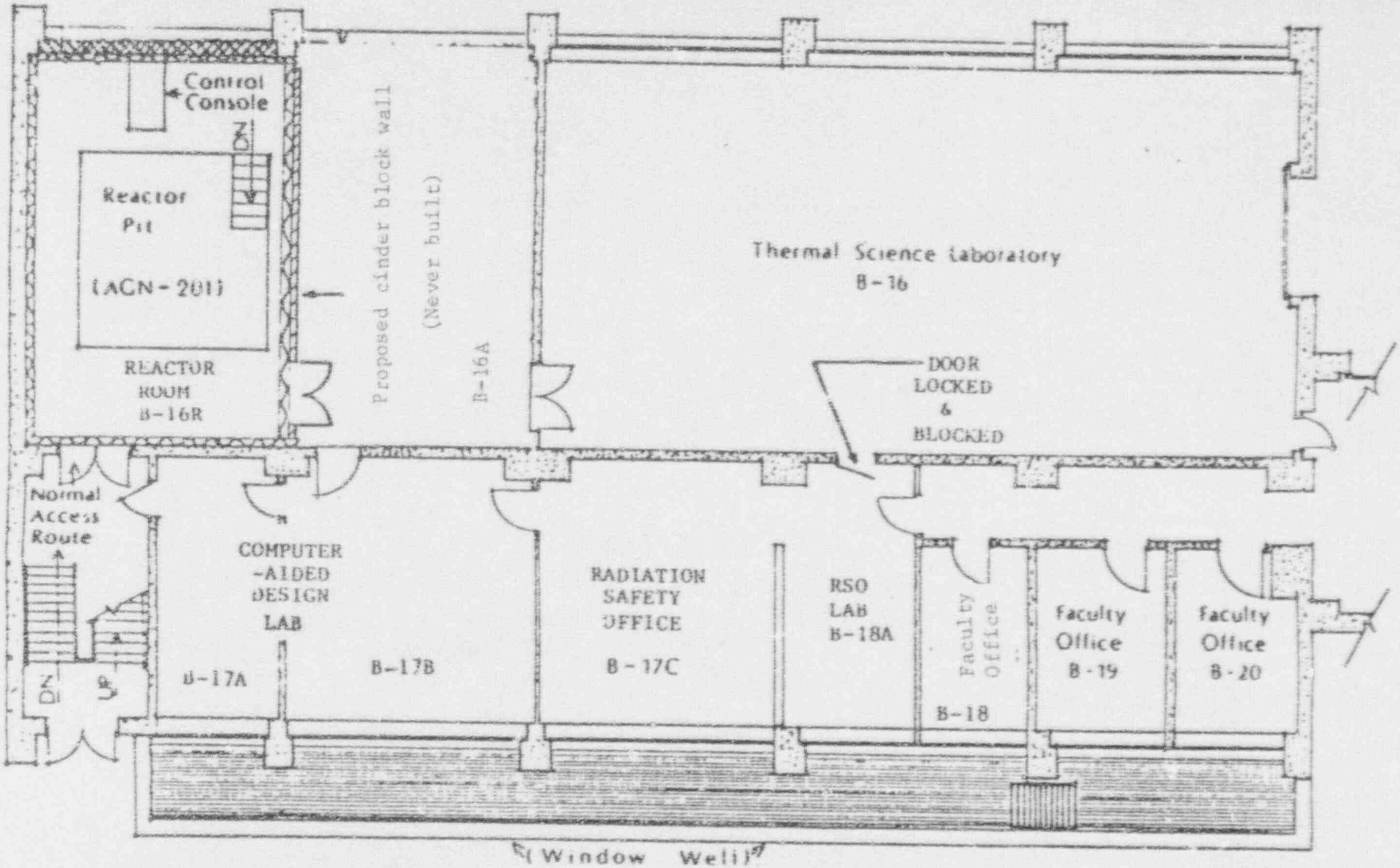


Fig. 1



THE CATHOLIC UNIVERSITY OF AMERICA

The Department of Mechanical Engineering

NUCLEAR REACTOR ROOM (B-16R PANGBORN HALL)

Fig. 2

5 0 5 10 15 20
scale feet

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Date: Dec 1991 (rev.)

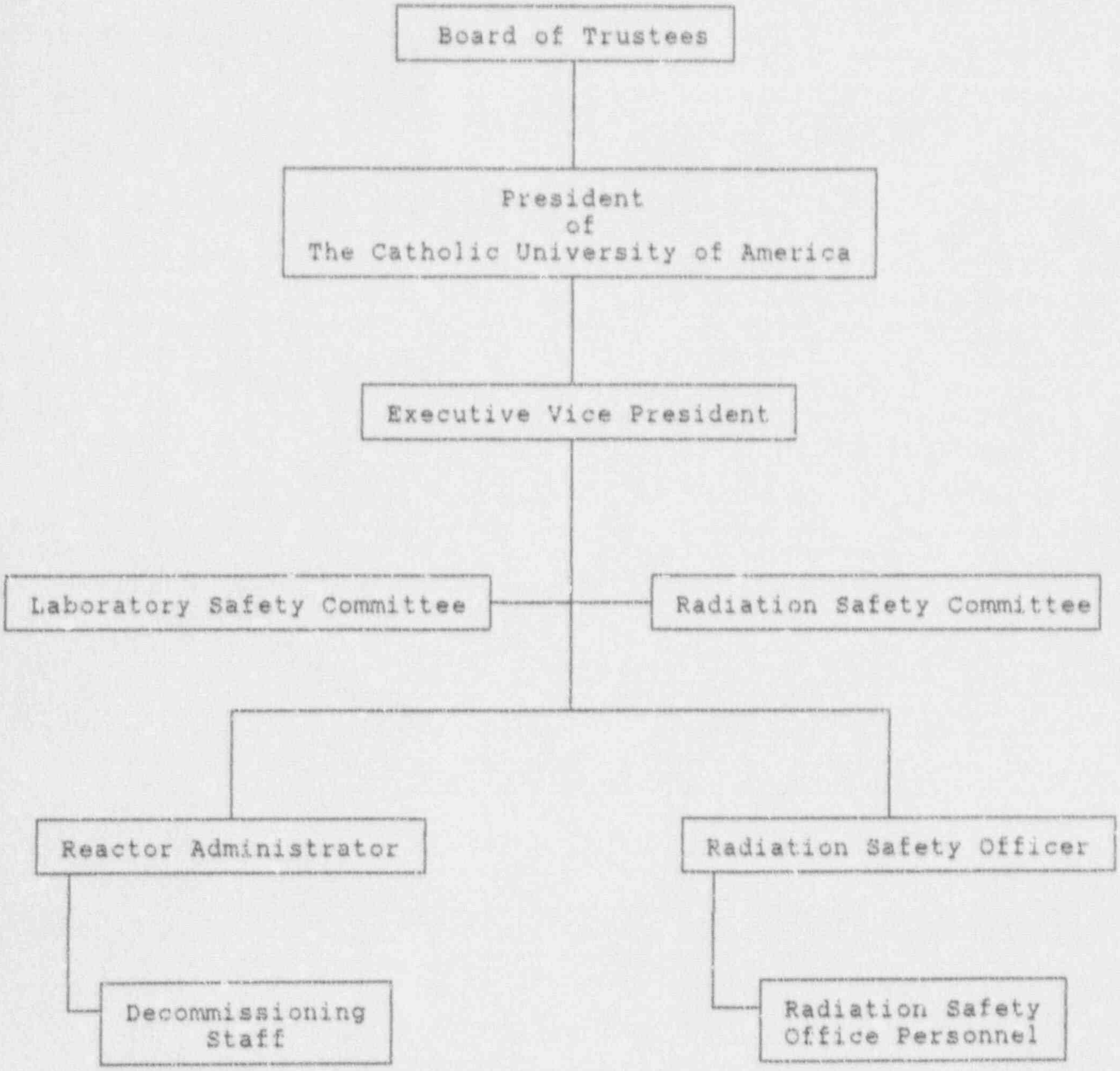

Figure 3. Administrative Organization for Decommissioning The Catholic University of America AGN-201 Nuclear Reactor

Appendix A: Description of AGN-201 Reactor

A.1 Introduction

The following description is excerpted from the Final Safety Analysis Report (FSAR) which was submitted in October 1977, in support of the license renewal application. The description applies to an intact, fueled reactor. It should be noted that, in the case of the CAGN, all fuel has been removed and disassembled, and the electrical cable which connected the console and the reactor has been removed and destroyed. A more complete description is found in Chapter 4 of the FSAR.

A.2 Description

The CUA AGN-201 Reactor, Serial No. 101, is a compact, portable, self-contained nuclear reactor designed to operate at a power level of 0.1 watt. The core, together with the fueled control and safety rods, contains approximately 682 g of U-235 in the form of 19.90% enriched UO_2 micro-particles embedded in radiation-stabilized polyethylene moderator. Shielding of graphite, lead and water surrounds the core. The maximum thermal flux is 5×10^6 n cm^{-2} s^{-1}. The complete reactor unit stands approximately 3 m (10 ft) high and 2.1 m (7 ft) in diameter. When so-licensed, continuous power levels as high as 20 w are possible with additional shielding.

The core is doubly sealed in an aluminum core tank and the steel reactor tank. A thermal fuse-link prevents deliberate high power operation.

The reactor system consists of two basic units, the Reactor and the Control Console. The reactor unit includes the uranium-polyethylene core, graphite reflector, and the lead and water shielding. Fuel-loaded control and safety rods are installed vertically from the bottom of the reactor unit. The weight of the reactor unit, with the water shield, is approximately 9,000 kg (20,000 lb); the weight of the console unit is 350 kg (800 lb). Power requirements are 2 kW of 100 VAC electrical power.

The critical mass necessary for operation of the Reactor is approximately 665 g of U-235. The uranium, in the form of UO_2 microspheres, is homogeneously embedded in approximately 10,900 g of polyethylene formed into a right circular cylindrical core 25.75 cm (10.15 in) in diameter and 23.75 cm (9.34 in) high.

The reactor is operated at sufficiently low power that, if all of the radioactive fission products formed in normal operation were to escape, they would not be dangerous to a person in the least advantageous position. Similarly, in the event of an excursion, no person would be seriously harmed under any plausible circumstances.

Appendix B: Summary of Reactor Operations 1957 - 1982.

THE CATHOLIC UNIVERSITY OF AMERICA

AGN-201, Serial No. 101

Initial Criticality: 20 November 1957
Final Shutdown: 4 December 1982
Reactor Defueled: 17 March 1990

Year	Hrs Operated	Energy Generated (watt-hours)
1957	5.4	.27
1958	12.8	.56
1959	14.1	.92
1960	25.9	.79
1961	13.1	.26
1962	19.1	.70
1963	2.4	.00
1964	10.4	.07
1965	16.2	.10
1966	32.5	.52
1967	29.6	.85
1968	46.6	1.52
1969	29.0	1.44
1970	45.5	1.43
1971	137.8	3.14
1972	10.9	.41
1973	56.1	1.09
1974	16.0	.34
1975	3.2	.03
1976	0	0
1977	0	0
1978	0	0
1979	17.6	.28
1980	36.9	.83
1981	22.1	.52
1982	6.6	.12
Total:	609.8 hours	16.19 watt-hours

Appendix C: Estimated Fission Product Inventory of CAGN Fuel

The fission product activity inventory of the CAGN fuel is not known. However it is possible to place an upper bound on it, using an estimating technique given by Cember[2].

Assuming a mean value of 190 MeV heat energy per fission, one watt is equivalent to 3.3 E10 fissions per second. It is estimated (App. B) that the total thermal energy released during all operation of the CAGN was 16.19 w h. Then, assuming all fissions to have been of U-235, the burnup of U-235 is estimated to be:

$$(3.3\ \mathrm{E10\ fiss\ s^{-1}}) * (16.19\ \mathrm{w\ h}) * (3600\ \mathrm{s\ h^{-1}}) = 1.9\ \mathrm{E15\ atoms},$$

which corresponds to about 750 nanograms of U-235.

This estimated burnup occurred over a period of about 25 years. To place an upper bound on the fission product activity at final shutdown (DEC 82), it is assumed that all fission product production occurred during the final operation. Then (Cember, op cit), the fission product activity at times up to 1000 hours (46.67 d) later is estimated from

$$A = 1.03\ \mathrm{E\text{-}16}\ T^{-1.2}\ \mathrm{Ci/fission,\ with}\ T\ \mathrm{in\ days},$$

and our upper bound for the fission product activity in mid-January 1983 would be approximated by

$$A = (1.03\mathrm{E\text{-}16\ Ci/fiss}) * (1.9\ \mathrm{E15\ fiss}) * (46.67)^{-1.2},\ \mathrm{or}$$

about 1.9 millicuries.

To estimate an upper bound for the current (DEC 91) fission product activity, use is made of information given by Eichholz[3]. In Table 164, Eichholz provides data relating the fission product activity of high level waste at 5 y to that at 1 y after reprocessing. From these values it can be inferred that the effective half life of the fission product activity during this period is about 1.9 y. Making the assumption that this value is reasonable over the nine years since last operation of the CAGN, our upper bound on the present fission product activity in the CAGN fuel is estimated to be

$$A = (1.9\ \mathrm{E\text{-}3\ Ci}) \exp(-9 \ln 2 / 1.9) = 7.1\ \mathrm{E\text{-}5\ Ci}.$$

Thus, it is estimated that the present fission product activity of the CAGN fuel is less than 71 microcuries.

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[2] Cember, Herman, Introduction to Health Physics, 2nd Ed., Pergamon Press, Elmsford, NY, 1983.

[3] Eichholz, Geoffrey G., Environmental Aspects of Nuclear Power, Ann Arbor Press, Ann Arbor, MI, 1977

Appendix D: Qualifications and Training of Decommissioning Personnel

WARREN E. KEENE

Position: Director of Radiation Safety, The Catholic University of America

Experience: 14 Years

Experience Summary:

Dr. Keene has held the position of Director of Radiation Safety (formerly titled Radiation Safety Officer) at The Catholic University of America since 1977. He has held numerous positions in the Baltimore-Washington Chapter of the Health Physics Society, serving as President during the 1987-88 Chapter year. For many years, he has been a lecturer in professional and technician level training courses conducted by the Chapter. In 1987 and 1988 he served as a member of the advisory panel to the Board of Directors of the Health Physics Society for the purpose of guiding the creation of a transition education program for health physicists, health physics technicians, and management personnel having corporate responsibility for compliance with radiation safety regulations. The program was intended to assist the target audiences in complying with anticipated changes to 10 CFR 20.

Specific Experience:

- Engaged in basic and applied research in radiation dosimetry and environmental radiation measurements 1966-1977.
- Engaged in basic and applied research in the application of porous-glass-based ion exchange techniques to the management of both low and high level radioactive waste 1977-1987. This work required the design and fabrication of special tools and equipment for hot cell use, as well as training hot cell operators in their use.
- Guided the development of an off-gas handling system for a laboratory scale slurry-fed melter used to produce uranium-thorium glass for radioactive waste management research in support of the reclamation of the West Valley, NY site.

Education:

Ph.D., Nuclear Science and Engineering,
The Catholic University of America, 1987
M.S., Nuclear Science and Engineering,
The Catholic University of America, 1966
B.S., Engineering Management,
The George Washington University, 1957

Appendix D: Qualifications and Training of Decommissioning Personnel

MOHAMMAD SADEGH SABA

Position: Health Physics Technician - I
The Catholic University of America
April 1991 - present

Experience:

March 1984-March 1986. Nuclear Power Plant Management, Atomic Energy Of Iran (AOEI). Reviewed nuclear design and modifications and the contract of nuclear power plant. Prepared sorting computer program for inventory of all the equipments and materials for the Iran Nuclear Power Plants.

October 1980-March 1984. Nuclear Research Center/Isfehan Technology Center, AEOI. Studied aspects of nuclear fuel design using computer codes such as PANTER, EREBUS and METHUSELAH. Designed a fuel lattice for the subcritical reactor in Isfehan.

September 1979-October 1980. Exxon Nuclear, Richland, WA. IAEA Fellow. Performed nuclear core analysis fuel management for BWR's and PWR's. Compared the results of XTG and XFYRE computer codes with General Electric codes. Performed the preliminary design of spent fuel assembly storage, using ORIGEN computer code. Used PDQ-7 to compare the results of point depletion and zone depletion methods.

September 1978-June 1979. Kraft Werk Union, Erlangen, Germany. Alongside Engineer in the fuel management department. Studied results of the MEDIUM computer code for different fuel strategies.

May 1976-September 1978. Nuclear Research Center, AEOI. Studied criticality and fuel burnup of Iran 1 & 2 Nuclear Power Plants using METHUSELAH and EQUIPOSE computer codes.

Education:

Bachelor of Science in Physics,
Pahlavi University, Iran June 1974

Special training in nuclear reactor technology, AEOI, April 1976. (Applied Mathematics, Nuclear Reactor Engineering, Nuclear Reactor Physics, Reactor Fuel Design, Heat Transfer, Computer Programming, and Numerical Analysis.

Graduate student, Joint Center for Graduate Studies, Richland, WA. Passed Nuclear Fuel Management and Nuclear Reactor Engineering courses, February 1980.

Graduate student (M.S. in Health Physics), Georgetown University, Washington DC, September 1991-present.

Appendix E: The CUA RSC Membership

Dr. Roland M. Nardone	Director of the Center for Advanced Training in Cell and Molecular Biology, and **RSC Chair**
Dr. Gregory rewer	Assistant Professor of Chemistry
Mr. Robert Fawbush	Manager, Electrical/Mechanical Services
Mr. Clay Goldston	Director of Public Safety
Dr. Jan Hallal	Associate Professor of Nursing
Dr. Warren E. Keene	Reactor Administrator and Director of Radiation Safety
Dr. Scott Keimig	Director of Environmental Safety
Dr. Isabelle Muller	Staff Scientist Vitreous State Laboratory
Dr. Daniel Sober	Professor of Physics

Appendix F. Reference Documents

1. Reactor License No. R-31

   U. S. Nuclear Regulatory Commission Docket No. 50-77.

2. ANSI/ANS-15.10-1981

   American National Standard for Decommissioning of Research Reactors

3. 10 CFR 20

   Standards for Protection Against Radiation

4. 10 CFR 30

   Rules of General Applicability to Domestic Licensing of Byproduct Material

5. 10 CFR 50

   Domestic Licensing of Production and Utilization Facilities

6. 10 CFR 71

   Packaging and Transportation of Radioactive Waste

7. 10 CFR 73

   Physical Protection of Plants and Materials

8. 49 CFR

   Department of Transportation Regulations Governing the Transportation of Radioactive Materials

9. Regulatory Guide 1.86

   U. S. Atomic Energy Commission, 1974, Termination of Operating Licenses for Nuclear Reactors

10. Radiation Safety Manual

    The Catholic University of America, 1980, as amended

APPENDICES G, H, & I ARE SEPARATELY BOUND

Appendix J: Abbreviations and Acronyms

AGN	Aerojet General Nucleonics
ALARA	As Low As Reasonably Achievable
CAGN	CUA AGN-201 Nuclear Reactor, Serial No. 101
CFR	Code of Federal Regulations
CRSO	CUA Radiation Safety Office
CUA	The Catholic University of America
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DRS	Director of Radiation Safety
EVP	Executive Vice President
NRC	U.S. Nuclear Regulatory Commission
ORO	DOE Oak Ridge Operations Office
RA	Reactor Administrator
RG	(NRC) Regulatory Guide
RSC	Radiation Safety Committee
RSO	Radiation Safety Officer
SNM	Special Nuclear Material