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PART II
SPECIFICATIONS VOLUME
SNM-696 MATERIAL LICENSE RENEWAL
GENERAL ATOMICS
SAN DIEGO SITE

Submitted November 1989

U. S. Nuclear Regulatory Commission
Docket No. 70-734

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FIGURES

TABLES

1. MATERIAL LICENSE LIMITS

General Atomic's San Diego facilities are located on two sites in close proximity to one another; its Main Site at 10955 John Jay Hopkins Drive and its Sorrento Valley Site at 11220 Flintkote Avenue and 3483 Dunhill Street. This license, SNM-696, authorizes General Atomics (GA) at its two San Diego sites, which includes facilities at 10955 John Jay Hopkins Drive, 11220 Flintkote Avenue, and 3483 Dunhill Street to acquire, deliver, receive, possess, use, and transfer special nuclear material in the forms of metal, metallic alloy, or compounds as follows:

Type	Kind	Quantity by Weight	
Uranium	Up to 19.99% enriched in U-235	200 kg U-235	
Uranium	20% to 100% enriched in U-235	< 5 kg	*
Uranium	Up to 100% enriched in U-233	< 2 kg	*
Plutonium	Encapsulated and/or sealed sources	< 2 kg	*
Plutonium	Plated calibration sources	5 gm	*
Plutonium	Bred but unseparated	1 kg	*
Plutonium	Solutions, precipitates & solids	5 gm	*

*The sum total quantity of strategic special nuclear material possessed at any one time must be less than 5,000 grams computed by the formula:

Grams = (grams U-235 contained in uranium enriched to 20% or more + 2.5 (grams U-233 + grams plutonium).

The foregoing quantities are the maximum that will be possessed under this license at any one time. No more than 1,000 gm of U-235 will be in the form of uranium hexafluoride.

This license also authorizes GA to possess in private carriage the above SNM, including any associated source or byproduct material in packages approved pursuant to 10 CFR 71 or otherwise in accordance with DOT regulation. Such material shall be protected during transit in accordance with 10 CFR 73, the licensee's approved security plans and conditions of license.

In addition to GA's Main Site (~60 acres) and its Sorrento Valley Site (~60 acres), this license extends to about 0.05 acres of non-GA land upon which is located GA's sewage pump house. This area, which is located immediately adjacent to GA's Main Site, has been referred to as Area B2 in correspondence to NRC, e.g., letter dated August 22, 1988.

2. AUTHORIZED ACTIVITIES

At its San Diego sites, the licensee may perform the following activities:

Receive, store, and transfer special nuclear material.

Conduct laboratory-scale physical, metallurgical, chemical, and engineering investigations utilizing special nuclear material, including pilot-scale process development.

Produce uranium-bearing fuel bodies.

Assemble such fuel bodies into HTGR, TRIGA, Thermionic or other type uranium reactor fuel elements.

Fabricate and assemble test elements, capsules, and other specimens.

Assemble plutonium bearing fuel bodies into thermoelectric power sources.

Reclaim and purify uranium-bearing scrap material resulting from the licensee's operations.

Concentrate and transfer waste and scrap containing special nuclear material.

Release liquid water generated in licensee's facilities to the municipal sewerage system.

Conduct Hot Cell-type operations on irradiated fuel elements and other fueled specimens.

Make changes, in accordance with subsection 3.4 "Work Authorization and Change", in facilities, equipment and/or procedures without license amendment unless the proposed change involves:

(1) A change in the license specifications or other related license conditions.

(2) A significant increase in radiation exposure of employees.

(3) An unreviewed safety question.

(4) A new fuel processing facility which involves ground breaking for a new building or

(5) A new production process. A new production process is one in which different feed materials are to be used whose safety has not been evaluated in the renewal application, may introduce a new element of safety (e.g., greater air distribution of one of the radioactive materials), or that may affect the physical or chemical properties of the radioactive bearing materials (e.g., different alloying materials).

An amendment application shall be submitted to NRC for new activities identified in 10 CFR 70.23, paragraphs 7 and 8, and activities covered by 10 CFR 51.5(b)(4). Examples of such new activity are plutonium fuel fabrication, commercial waste disposal or conversion of UF_6 in production quantities greater than 1 kg U-235.

In addition the licensee is authorized to possess SNM and its associated source and byproduct material in private carriage within the United States between GA's licensed facilities, and to/from GA's licensed facilities to/from persons authorized to receive such materials, and from a carrier's terminal to GA licensed facilities. The conduct of the aforementioned activities shall be in accordance with GA's NRC-approved Fixed Site and Transportation Plan for the Protection of Special Nuclear Material of Moderate and Low Strategic Significance (See Section 8).

3. ORGANIZATION AND ADMINISTRATIVE PROCEDURES

3.1 GENERAL

This section describes the organizations, positions, and administrative procedures utilized to control the use of special nuclear material (see Fig. II 3.1-1).

The licensee may change organizational responsibilities, reporting locations and names providing such changes do not adversely affect the implementation of license conditions and are reported to the NRC within 60 days after the change has been made.

3.2 RESPONSIBLE ORGANIZATIONS

3.2.1 Operating Groups

All operating groups are required by company policy to conduct their respective activities within federal, state, and local rules and regulations, license criteria, and company policy, criteria and established practices. Such policy requires operating groups to conduct their operations so that radiation exposures to employees and the general public are as low as reasonably achievable.

The responsibility for safe operation and control of activities in a specific SNM or restricted area and for the safety of the environs as influenced by the activities conducted therein shall be vested in the manager of each facility or his or her designee. The manager or designee shall ensure that the conduct of activities within the area is in compliance with all applicable criteria, rules and practices as set forth in Work Authorizations issued under this license implementing the

Commission's regulations. By continuous supervision, the manager can take any corrective action necessary to assure license compliance.

3.2.2 Compliance Functions

All functions responsible for assuring compliance with applicable license requirements and controlling the radiological and nuclear safety and safeguards of licensed material are part of the Human Resources organization of General Atomics. Namely, these functions are: Nuclear Safety, Licensing, Safety and Nuclear Compliance, Nuclear Material Accountability, Statistics & Measurement Control, Security, and Health Physics.

The Vice President of Human Resources, or his designee, will establish the necessary policies of operation, cause them to be published in company-wide guides and manuals, and coordinate related activities with operating groups to assure compliance with related policies, procedures, regulations and license conditions. The Vice President of Human Resources reports directly to a member of the Board of Directors of General Atomics (see Figure II 3.1-1).

The compliance functions are described below and are headed by managers or supervisors which are synonymous titles for signifying the responsible person.

3.2.2.1 Licensing, Safety and Nuclear Compliance (LSNC)

This function administers licenses and reviews and approves all Work Authorizations (WA) involving SNM for compliance to applicable regulation and license conditions. This function provides interpretation of licenses and regulations, determines the need for licensing actions, coordinates the preparation and

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3.2.2.4 Nuclear Materials Accountability (NMA)

This function assures compliance with SNM custody and control rules, regulations and practices. This function implements the program for accountability, custody and control of special nuclear material. NMA maintains a manual of SNM accounting control procedures. NMA assures compliance with safeguards material control and accountability regulations and license conditions.

3.2.2.5 Statistics & Measurement Control (SMC)

This function is responsible for implementing the measurement control program as it relates to SNM control and accountability. This function assures SNM measurement quality by monitoring and evaluating applicable information, practices and procedures. This function is also responsible for maintaining a SNM Measurements & Statistical Control Manual setting forth the procedures for implementation of measurements, measurement control and statistical control requirements, and implements the program for furnishing measurement and statistical information for inventories, receipts, and shipments.

3.2.2.6 Security

This function provides security measures to prevent purposes, of proposed SNM activities, reviews proposed changes in processes, equipment, and procedures, and performs frequent inspection and monitoring to assure adequate implementation of nuclear safety controls. This function approves for adequacy all determinations of criticality limits and provides and/or obtains independent verification thereof as well as approves the content of training programs in nuclear criticality safety.

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3.3.1 Facility Managers

A Facility Manager shall have demonstrated his/her proficiency in activities relevant to the functions assigned, as defined in paragraph 3.2.1. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree, a minimum of two years applicable work experience and/or training in nuclear activities, and knowledge of the licensee's radiation protection program.

3.3.2 Compliance Functions

The Vice President of Human Resources is responsible for assuring compliance with applicable licensing requirements and controlling the radiological and nuclear safety and safeguards of licensed material, as generally defined in Paragraph 3.2.2. Incumbents must have a minimum of five years of nuclear industry management experience of a high level general management nature.

The Manager, Licensing, Safety and Nuclear Compliance shall have demonstrated his/her proficiency in activities relevant to the functions assigned, as defined in Paragraph 3.2.2.1. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering or other related field, a

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accountability and control activities.

The Manager, Statistics and Measurement Control shall have demonstrated his/her proficiency in activities relevant to the functions assigned, as defined in paragraph 3.2.2.5. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering or other related field and a minimum of two years applicable work experience and/or training in quality assurance/control, statistics, nuclear measurements, or related activities.

The Manager, Security, shall have demonstrated his/her proficiency in activities relevant to the functions assigned, as defined in paragraph 3.2.2.6. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year degree and a minimum of two years applicable work experience and/or training in managing or implementing security programs in the nuclear industry.

A Health Physics Technician shall have demonstrated his/her proficiency in activities relevant to the functions assigned, as described in Section 3.2.2.2. Demonstration of this proficiency shall be based on the technician's qualifications which shall include a high school diploma or equivalent, and a minimum of two years applicable work experience and/or training related to nuclear activities, including at least three (3) weeks of specific on the job training.

3.3.3 <u>CRSC</u>

Members of the Criticality and Radiation Safety Committee (CRSC) shall have expertise in their special fields. Such expertise shall, as a minimum for technical members, require an accountability and control activities.

The Manager, Statistics and Measurement Control shall have demonstrated his/her proficiency in activities relevant to the functions assigned, as defined in paragraph 3.2.2.5. Demonstration of this proficiency shall be based on the manager's qualifications which shall include an accredited four-year college degree in science, engineering or other related field and a minimum of two years applicable work experience and/or training in quality assurance/control, statistics, nuclear measurements, or related activities.

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3.3.3 CRSC

Members of the Criticality and Radiation Safety Committee (CRSC) shall have expertise in their special fields. Such expertise shall, as a minimum for technical members, require an accredited college degree or equivalent training in a field appropriate to their specialties and a minimum of two years related experience. For nontechnical members, a minimum of three years experience in an appropriate field is required.

The fields of specialty shall include as a minimum, disciplines from nuclear physics, reactor physics, health physics, chemistry and statistics. A member may represent more than one field of specialty.

3.3.4 Independent Nuclear Safety Reviewers

Employees performing independent nuclear safety analysis and review shall have their qualifications reviewed and approved by the manager of the reviewing organization and the Manager of Nuclear Safety or the Chairman, CRSC. Each of the persons performing such analysis or review or portion thereof may have their relevant experience in a different specific phase, but this will allow broader expertise to be utilized.

3.3.5 Training

Radiation safety training and indoctrination shall be conducted by the Health Physics Manager or by a similarly qualified individual approved by the Health Physics Manager. The Health Physics Manager may delegate training in that portion of the course to an individual who is uniquely qualified to present it.

New Employees - Radiation safety training, appropriate to the employee's needs, shall be given to all new employees.

Authorized Users - Employees who handle SNM other than under the direct supervision of another trained individual, must have satisfactorily completed the Radiological Safety Course which includes the following topics: ALARA practices, an introduction to 10 CFR 19 and 10 CFR 20, radiological instrumentation and controls, decontamination procedures, and emergency procedures. General subjects such as the nature and sources of radiation, units, methods of controlling contamination, interactions of radiation and matter, biological effects of radiation, use of monitoring equipment, principles of nuclear criticality safety and risks from occupational radiation exposure are also covered. The course is presented by lectures and demonstrations including the use of selected audiovisual aids.

The understanding of the employees and the effectiveness of the training course are evaluated with an examination given at the end of the course. Employees are required to pass the written examination with a grade of 70 percent before being allowed to handle radioactive material without direct supervision. The exam includes radiological practices and nuclear safety principles. The Manager, Health Physics, or other qualified designee is responsible for the radiological safety training course content.

Retraining - Retraining on selected subjects shall be conducted periodically. For employees working with more than 15 grams of U-235, retraining shall be conducted every two years. For employees working with 15 or fewer grams of U-235, the retraining shall be conducted every three years. Subjects will include both radiological and nuclear safety controls. The understanding of the employees and the program's effectiveness will be evaluated with an examination which must be satisfactorily completed with a score of 70 percent. Records of the training will be kept which will include the date held, subject matter covered, attendees, instruction and test results. The Manager, Health Physics or other qualified designee is responsible for this training.

Employees who are NRC-licensed reactor operators or senior reactor operators are exempt from the retraining requirements of this license.

Health Physics Technician Training - All Health Physics Technicians shall successfully complete the Radiological Safety Course described above and receive on-the-job training. Additional Nuclear Safety Training shall be given to enable them to do nuclear safety monitoring on a "continuous basis" in areas where more than 500 grams of U-235 is handled. Training will include topics such as characteristics of criticality, principles of nuclear safety, specific SNM station limits and justifying criteria and details of postulated criticality accidents.

Records of training include the date held, subject matter covered, attendees, instructor and test results. Such records will be kept for two years or longer if so required by Part 19 or 20.

3.4 WORK AUTHORIZATIONS

Any operating group desiring to initiate new work or to make changes in previously approved work involving special nuclear material subject to this license may do so only upon receipt of proper authorization. Procedural requirements and instructions for obtaining Work Authorizations (WA) are to be maintained in current form in the Radiological Safety Guide. The basic procedure for obtaining approval of a Work Authorization is described in section 3.4.1 and is shown schematically in Figure II 3.4.1. Changes to this procedure require review and approval by the Manager of Licensing, Safety and Nuclear Compliance, the Vice President of Human Resources, and the Chairman of CRSC, or their designees. All such changes must be within limitations of this license.

The criteria for WA approval are contained in Section 3.4.2.

3.4.1 Procedure for Approval of Work Authorizations

1. The operating group desiring to do the work shall prepare a Work Authorization (WA) request identifying the proposed work and quantity and form of SNM to be used as well as drafts of the necessary safety related procedures, equipment features, and process characteristics. The WA shall be approved in writing by the applicable manager who must assert that all license and company criteria and procedures for radiological safety, criticality, material accountability and control, and physical protection requirements will be met. Also, he must assert that other applicable safety related features of the work, such as structural integrity, potential of fire or explosion and the like, are adequately considered and suitable provisions have been incorporated. The approved WA and backup information shall then be forwarded to Health Physics.

2. The WA and appropriate safety related information shall be approved in writing by Health Physics, Nuclear Safety, Nuclear Material Accountability, and Licensing, Safety and Nuclear Compliance. In addition, the request shall be sent to CRSC for approval if the WA is for new or changed work involving new or revised nuclear safety analysis and involves a quantity of SNM greater than 350 gm of U-235, 200 gm of U-233, 200 gm of encapsulated plutonium, or any quantity of unencapsulated plutonium of any isotope. Security will be consulted and concurrence obtaining if physical

protection requirements for the area are to be modified. If a license modification is required, Licensing, Safety and Nuclear Compliance shall make the determination and take appropriate action with governmental regulatory authorities. Any reviewer shall seek outside expertise as deemed necessary.

3. After the WA has been approved in writing, authorization shall be deemed granted. Authorization shall include any special conditions required by the license and such other conditions as the reviewers deem necessary.

4. The originating department shall revise, where appropriate, the operating procedures and practices before implementing requested work. The current version of Work Authorization and the approved procedures will be available in or near the applicable work area.

5. All SNM Work Authorizations will be re-reviewed and approvals updated at least every three years.

6. Work Authorization records including supporting criticality analysis shall be retained for at least six months following the expiration date of the authorization.

3.4.2 CRITERIA FOR WA APPROVAL

Evaluation of each request for WA approval shall include but not be limited to the following determinations by the manager, or designee, of the respective responsible compliance functions:

1. That the proposed work is within currently authorized license limits.

2. That radiological safety is properly assured. This includes a review of operational methods to attain "as low as reasonably achievable" radiation exposures.

3. That criticality safety has been reviewed and adequate provisions are incorporated.

4. That new or revised nuclear safety analysis has been separately verified by two independent parties. Analysis of solid angles of interaction require verification by a single independent party.

5. That proper SNM accountability control is provided.

6. That proper material protection for SNM is provided and that applicable limits and regulations are observed.

7. That an assessment of the effluents has been performed and proper provisions for control exist.

8. That industrial safety practices and procedures are considered as they relate to radiological and nuclear safety.

3.5 <u>RADIOLOGICAL WORK PERMIT</u>

The Radiological Work Permit (RWP) is intended primarily for authorization of temporarily assigned GA employees who are not on a Work Authorization, or outside contractors or subcontractor personnel who will perform limited or routine work in a Controlled Area, but whose exposure to possible radiation or contamination is incidental to their function.

Radiological Work Permits (RWP's) shall be issued for all unplanned or non-routine work with licensed material not covered by a WA. The RWP shall be signed by Health Physics management or a senior staff member before related work can commence. A senior staff member shall be a Health Physics Technician having a minimum of two years' experience in radiation safety. An evaluation of the safety effectiveness of the permit shall be made upon completion of the work.

Any work performed on equipment which handled radioactive materials shall be authorized by and performed under a radiological work permit or approved Work Authorization.

3.6 INTERNAL INSPECTION AND AUDIT

3.6.1 Inspections and Reviews

Inspections and reviews of activities involving materials subject to this license are performed by Health Physics, Nuclear Safety, Nuclear Material Accountability, and Security. A Health Physics semiannual inspection[1] of SNM activities involving the handling of more than 15 grams of U-235 subject to this license shall be made by the Manager, Health Physics or qualified designee. The Manager, Nuclear Safety, or his designee performs criticality inspections[1] of all areas possessing SNM as follows: at least once annually for all areas and additionally whenever a Material Balance Area's throughput exceeds 3.5 formula kilograms. The NMA inspections and reviews are conducted in accordance with

[1]These shall be conducted according to a written plan which will include, but not be limited to, the scope of identified discrepancies, unusual events, corrective actions, review of operations involving any substantially new activity.

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control is being properly used, maintained, and inspected. The CRSC evaluation, recommendations, and corrective actions shall be documented and the report shall be sent to the appropriate operations managers, the Manager of Licensing, Safety and Nuclear Compliance and the Vice President of Human Resources.

3.7 OPERATING POLICIES AND PROCEDURES

3.7.1 General

Policies applicable to work with materials subject to this license shall be approved by the Vice President of Human Resources, or his designee, and as applicable, receive prior review by CRSC. Design criteria applicable to work with materials subject to this license shall be approved by the Manager, Licensing, Safety and Nuclear Compliance, and the Manager, Nuclear Safety, or their designees, and as applicable receive prior review by CRSC. Any operating department may have additional and more stringent criteria or procedures providing such procedures are within approved policies and criteria.

3.7.2 Activities Requiring Procedures

Procedures are required for all activities in which materials subject to this license are physically handled, stored and chemically or physically changed.

3.7.3 Unusual Events

Any unusual event which significantly threatens or lessens the effectiveness of radiological or health safety provisions of the license will be investigated and reported to the Vice President of Human Resources, and the applicable vice presidents. The Vice President of Human Resources or his designee is control is being properly used, maintained, and inspected. The CRSC evaluation, recommendations, and corrective actions shall be documented and the report shall be sent to the appropriate operations managers, the Manager of Licensing, Safety and Nuclear Compliance and the Vice President of Human Resources.

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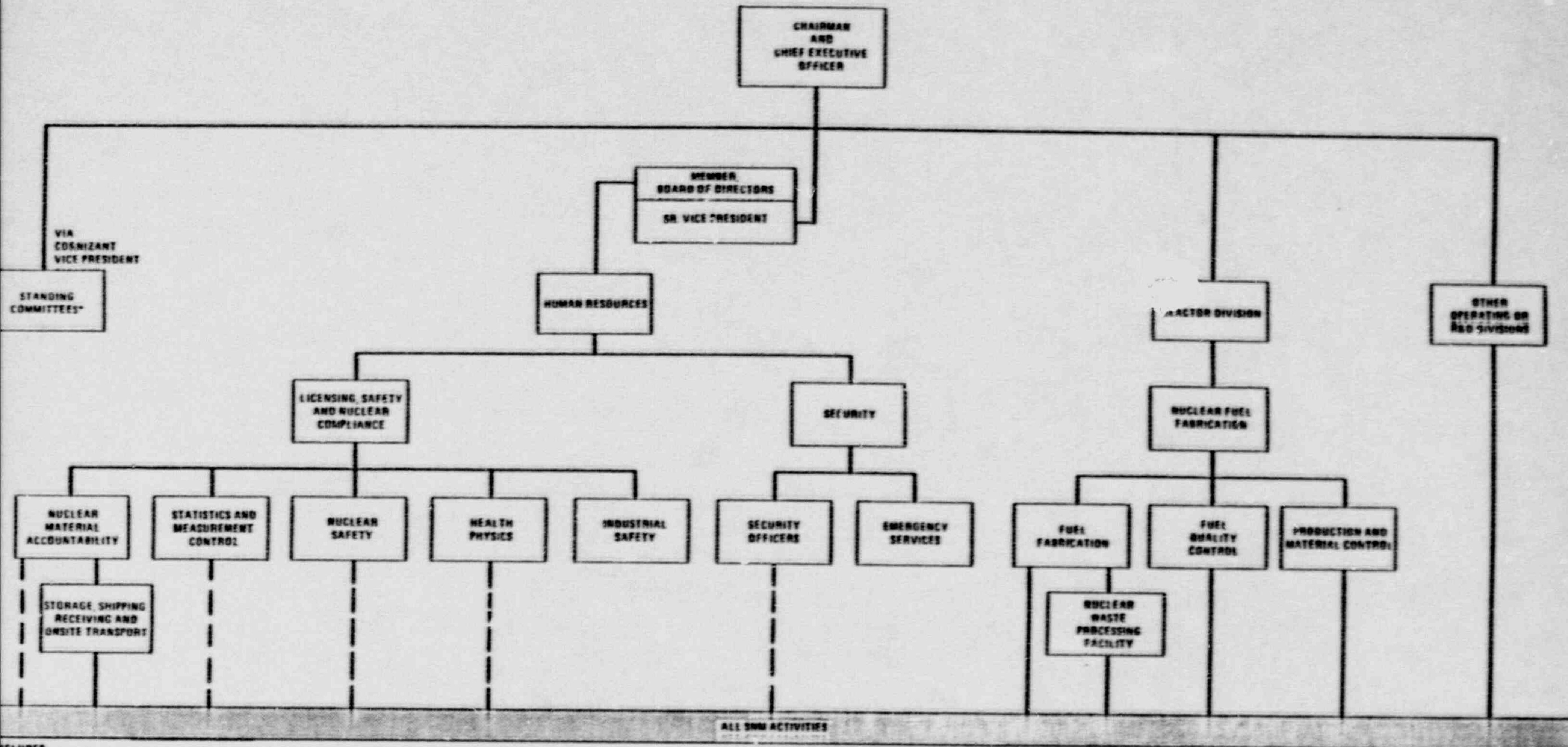

NCLUDES:
ITICALITY AND RADIATION SAFETY COMMITTEE
FEGUARDS MANAGEMENT REVIEW COMMITTEE
CURITY MANAGEMENT AUDIT COMMITTEE
GA CRITICALITY SAFETY COMMITTEE

REVISED MARCH 1986

FIG. II 3.1-1. Radioactive Materials Management Organization Chart

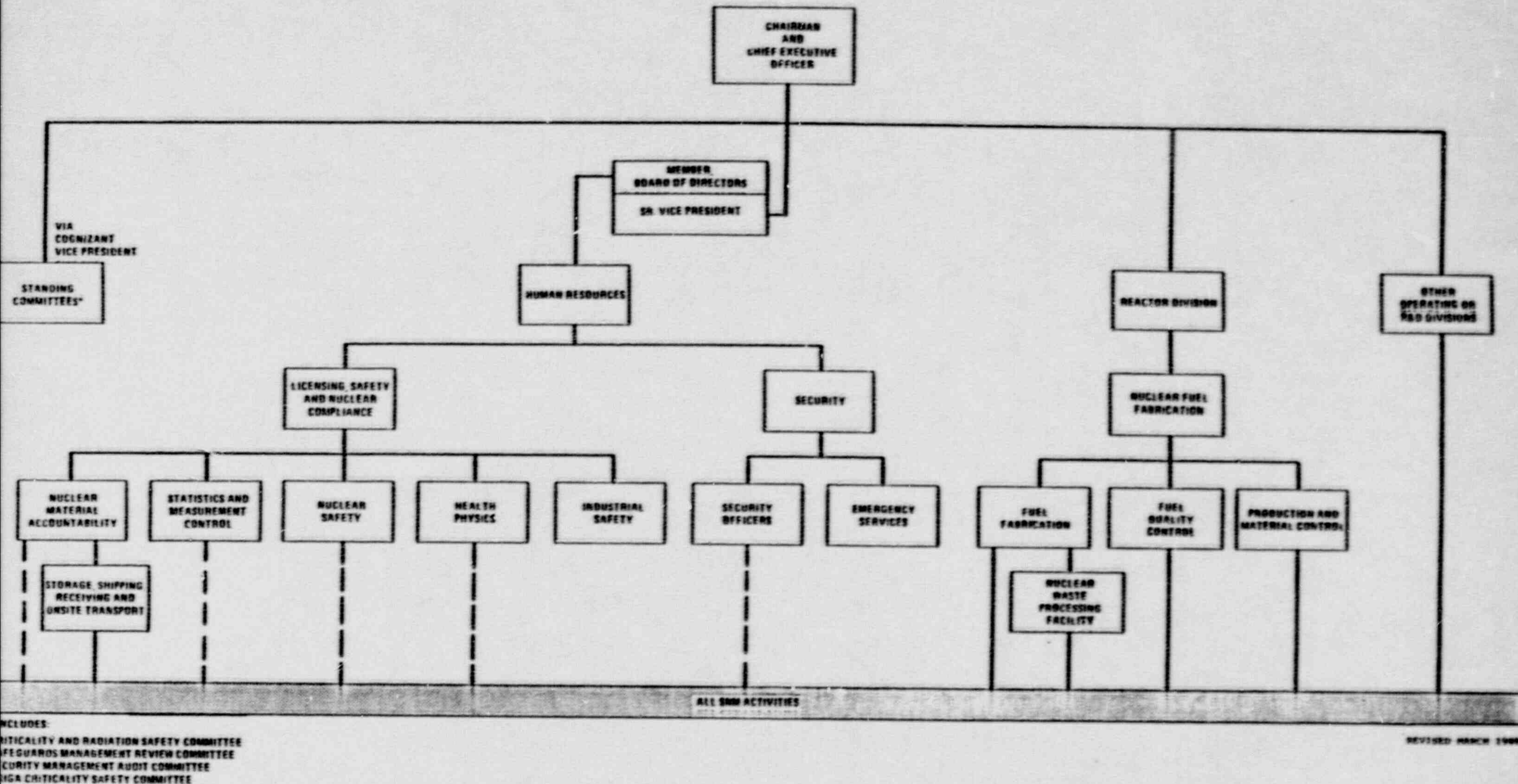

FIG. II 3.1-1. Radioactive Materials Management Organization Chart

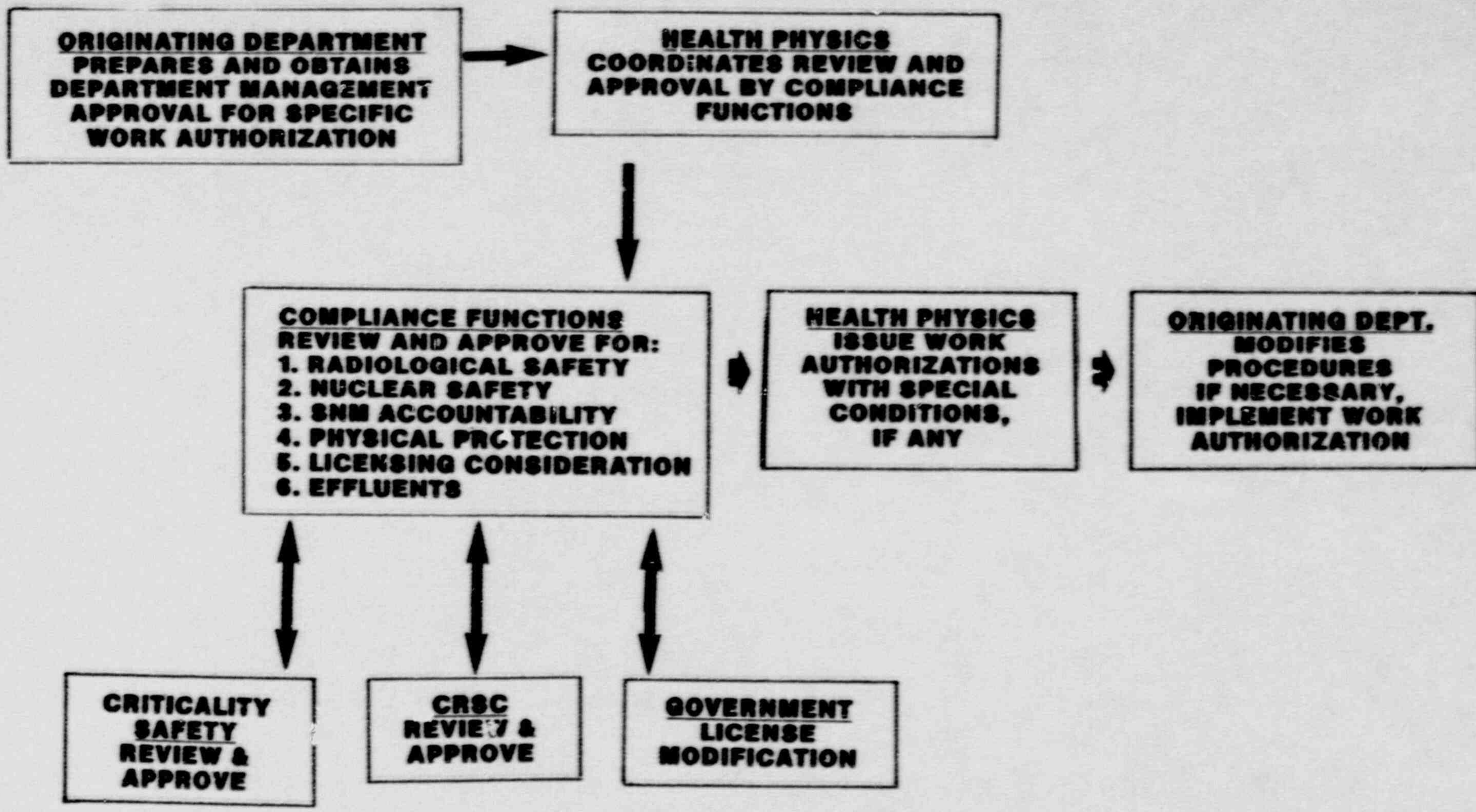

FIG. II 3.4-1 PROCEDURE FOR AUTHORIZING WORK

4. RADIOLOGICAL SAFETY

4.1 PERSONNEL EXPOSURE CONTROL

4.1.1 Controlled Access

Access to restricted areas shall be controlled for purposes of radiation safety. Posting of areas within a restricted area shall be in compliance with 10 CFR Part 20.

4.1.2 Work Places

Unsealed and unencapsulated uranium or unencapsulated and bred but unseparated plutonium shall be handled only in designated work places. Work places are classified as Type I, II or III depending on degree of protection required with a Type I work place having the lowest exposure potential. Table II 4.1-1 gives the allowable quantities of materials for each type of work place.

Work places shall have the following minimum requirements.

4.1.2.1 Type I

1. Room ventilation shall provide at least four air changes per hour (except in areas when all material is stored and equipment is cleaned out).
2. The work surfaces shall be smooth and impermeable.
3. Personnel shall wear appropriate distinctively marked protective clothing.
4. The radioactive material shall be stored in appropriate and properly labeled containment when not in use.
5. A periodic monitoring program shall be maintained to detect surface contamination, unusual sources of radiation, and airborne radioactivity.

Table II 4.1-1

WORK PLACE MATERIAL ALLOWABLE QUANTITIES

Material Form	Type I Pu	Type I U	Type II Pu	Type II U
Gases	N/A	None	N/A	None
Material in a form which can become airborne	None	None	None	(1)
Wet Chemical Operations	None	5 gm	1 gm	(1)
Coated particles and other forms where there is no possibility of material becoming airborne	None	(1)	0.5 gm	(1)
Storage	Storage requirements based on radiological and criticality considerations			

(1) Limit based on criticality considerations.

Type III work places are for use with radioactive materials in quantities greater than those specified above for Type II work places. Type I and Type II work places may be contained within Type III work places.

4.1.2.2 Type II

Work places which have amounts of radioactive material in excess of those stated for Type I work places shall meet the following additional requirements:

1. Operations involving materials in a form which can become airborne shall be carried out in enclosures or exhaust ventilation systems having a face velocity of 150 ft per minute over 90% of the opening. The enclosure or systems shall be equipped with high-efficiency particulate air filters as appropriate to the operation being performed. Work shall be halted if the average face velocity falls below 100 ft/min. Such operations shall be monitored with air samplers in accordance with Section 4.1.4.2.

2. Hands shall be checked for contamination at appropriate stages during operations and prior to leaving the work place.

3. Receptacles shall be provided for separate collection of solid and liquid residues generated in the work place.

4.1.2.3 Type III

Type III work places meet the following requirements in addition to those listed for Type I and II work places.

1. Operations shall be carried out in inert boxes, gloved boxes, closed process equipment or equivalent enclosure* equipped with negative pressure ventilation and high-efficiency particulate air filters.

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*"Equivalent enclosure" to be determined during Work Authorization review.

Other protective devices shall be included commensurate with the degree of hazard associated with the operations; namely, shielding, remote handling devices, air locks, bag-out ports, etc.

2. The atmosphere in work rooms shall be kept at negative pressure with respect to other areas of the building.

3. Room exhaust shall be filtered through high-efficiency particulate air filters.

4. An air sampling program shall be maintained in accordance with Section 4.1.4.2. If air sample results indicate an increase in airborne contamination to 10 percent of the appropriate MPC, survey frequencies shall be increased to the levels specified for Type II laboratories.

4.1.3 Personnel Monitoring

Personnel shall be monitored in accordance with 10 CFR Part 20.202 utilizing equipment described in Section 4.2. In addition, a bioassay program shall be conducted for employees working with unencapsulated or unsealed SNM. This program shall be used in any area where air sampling is necessary for personnel protection and shall use urinalysis as a screening procedure and utilize in vivo counting techniques to determine the magnitude of an individual's dose resulting from inhalation or ingestion of SNM.

Bioassay Type & Frequency Criteria

An individual shall submit specimens for urinalysis prior to commencement of work in such an area described above. Such individuals shall be counted on the first scheduled visit of the in vivo measurement equipment.

Interpretation of Bioassay Results (action levels and actions)

Appropriate action as based on bioassay results is dependent first on the underlying purpose of the measurement.

1. Preparatory Evaluation _ An investigation will be conducted if the presence of any urinary uranium, as detected by urinalysis procedures, has been found in personnel prior to job assignment.

2. Exposure Control - When work is in progress and bioassay measurements made routinely, the results will be reviewed and action taken as follows:

    a. Any "urinalysis" result which exceeds 20% of the Maximum Permissible Limit (MPL) will be reviewed and investigated. Appropriate corrective action will be taken as required. Additional bioassay "urinalysis" measurements will be obtained as required.

    If the result is >100% of the MPL, and unexpected, work restrictions will be established for all affected employees and individual case studies for affected employees will be performed.

    b. Any in vivo "U-235 lung count" measurement which exceeds 40% of the Maximum Permissible Lung Burden (MPLB) will be reviewed and investigated. Appropriate action will be taken as required.

    If the result is >100% of the MPLB, work restrictions will be established for all affected employees and individual case studies for affected employees will be performed. Contamination confinement and/or air sampling capabilities will be reviewed and corrected, unless a result of this magnitude was expected because of a known unusual cause and corrective action has been or is being taken.

4.1.4 Surveillance

4.1.4.1 Radiation and Contamination Surveys

Radiation and contamination surveys of restricted areas shall be performed as required by 10 CFR 20.201.

The frequency of measurement for contamination levels shall be as described in Table II 4.1-2.

TABLE II 4.1-2
FREQUENCY OF MEASUREMENT FOR
CONTAMINATION LEVELS

Restricted Area	Time
Lunch, Change and Rest Rooms	Weekly
Type I Work Place	Monthly
Type II Work Place	Weekly
Type III Work Place	Daily

Air sampler placement shall conservatively represent the workers breathing zone. The location of air samplers shall be checked annually and whenever any process or equipment changes are made to verify the representativeness of work area air sampling. In addition, the location of air samplers shall be checked at the commencement of operations in any area that has been shutdown for more than 6 months to verify the representativeness of air sampling. Each of the air samplers with suitable collection filters shall be coupled to a measuring device to determine the volume of air that flows through the filter. The collection filter papers shall be evaluated for alpha, beta, and gamma activity as appropriate to the operation.

Samples shall be collected for each work shift (usually eight hours) at the TRIGA Fuel Fabrication Facility. Samples shall be collected weekly in other areas such as radiochemistry and pilot-plant facility (SV-B), unless airborne releases are expected due to an incident, ventilation problem, etc. Samples shall be collected monthly in areas where there is reduced activity i.e. shutdown locations in fuel fabrication areas. If air sample results indicate an increase in airborne contamination to 10 percent of the appropriate MPC, survey frequencies shall be increased to the levels specified for Type II laboratories. The samples will be analyzed typically within three (3) days.

Any one sample measuring $\geq$1 MPC shall be reported to the Manager, Health Physics and investigated, and, if applicable, corrective action shall be taken.

Where air sample data indicates airborne radioactivity in excess of 50% of the MPC for U-235 in a specific area or location averaged over one week, the results shall be investigated and corrective action taken, as required.

The laboratories in which plutonium in a dispersible form may be used shall have exhaust ventilation systems separate from other building exhausts and shall provide dual High Efficiency Particulate Air (HEPA) filtering of the effluent air.

4.1.4.3 Water

Liquids from effluent waste streams from facilities utilizing SNM are collected in holdup tanks or drums and shall be grab sampled and analyzed for radioactivity before release to the sewerage system. Upon determination that the levels will be in accordance with 10 CFR 20.303, the contents are released to the sewerage system. All such releases are recorded.

Sewage shall be continuously grab sampled, samples collected daily during the normal work week, and evaluated for gross-alpha and gross-beta concentrations daily.

Investigation - Anytime the data indicate that the level averaged over one quarter exceeds 50% of the MPC, an investigation shall be made to determine the corrective action required, if applicable.

Intervention - If the concentration levels exceed the permissible levels, the water from the hold-up tanks is not released to the sewer, it is diluted by additional effluent makeup or treated as radioactive waste. Contents of the hold-up tank are annually sparged to prevent accumulation of material at the bottom of the tank.

4.1.5 Sealed Plutonium Source Leak Testing

Each sealed plutonium source shall be tested for leakage at intervals not to exceed 6 months. In the absence of a certificate from a transferrer indicating that a test has been made within 6 months prior to the transfer, the sealed source shall not be put into use until tested.

The test shall be capable of detecting the presence of 0.005 microcuries of alpha contamination on the test sample. The test sample shall be taken from the source or from nearest accessible surfaces of the device in which the sealed source is permanently or semipermanently mounted or stored. Records of leak test results shall be kept in units of Ci and maintained for inspection.

If the test reveals the presence of 0.005 microcuries or more of removable alpha contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it to be decontaminated and repaired by a person appropriately licensed to make such repairs, or disposed of in accordance with the governmental regulations. Within 5 days after determining that any source has leaked, the licensee shall file a report with the Nuclear Regulatory Commission, describing the source, the test results, the extent of contamination, the apparent or suspected cause of source failure, and the corrective action taken. A copy of the report shall be sent to the Director of the nearest Regional Office listed in Appendix D of Title 10, Code of Federal Regulations, Part 20.

The periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources exempted from this test shall be tested for leakage prior to any use or transfer to another person unless they have been leak tested within 6 months prior to the date of use or transfer.

Leak testing of sealed plutonium sources and alpha sources shall be in accordance with Annex A, "License Condition for Leak Testing Sealed Plutonium Sources," dated November 1979, and Annex B, "License Condition for Plutonium Alpha Sources," respectively.

4.1.6 Respiratory Protection

Respiratory protective equipment may be used in circumstances where the airborne concentration limits of 10 CFR 20.203(d)(1)(ii) cannot be practically achieved through the use of process or engineering controls or exposures controlled through other precautionary procedures such as increased surveillance, limitation of working times, etc. Any such use of respiratory protective equipment shall be conditioned upon the following:

1. The selected respiratory protective equipment will provide a protection factor greater than the multiple by which peak concentrations of radioactive materials are expected to exceed the values specified in Appendix B, Table I, Column 1 of 10 CFR Part 20. The equipment selected is used so that the average concentration of radioactive material inhaled during any period of uninterrupted use in an airborne radioactivity area, on any day, by any individual using the equipment, will not exceed the values specified in Appendix B, Table I, Column 1 of 10 CFR Part 20.

2. Each respirator user(s) is advised that he/she may leave the area for relief from respirator use in the event of equipment malfunction, physical or psychological discomfort, procedural or communication failure, significant deterioration of operating conditions, or any other condition that might cause reduction in the protection afforded the wearer.

3. The licensee maintains a respiratory protective program adequate to assure that the objective of item 1 above is met and such program shall include:

a. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposure, and to permit proper selection of the respiratory protective equipment.

b. Procedures to assure proper selection, supervision and adequate training of personnel using such protective equipment.

c. Procedures to assure the adequate fitting of respirators and the testing of equipment for operability.

d. Procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

e. Implementation of written operational and administrative procedures for control, issuance, proper use, and return of respiratory protective equipment, including provisions for planned limitations on duration of respirator use for any individual as necessitated by operational conditions.

f. Bioassays and other surveys, as appropriate, to evaluate individuals exposures and to assess protection actually provided.

g. Records sufficient to permit periodic evaluation of the adequacy of the respiratory protective program.

h. Determination prior to assignment of any individual to tasks requiring the use of respirators that such an individual is physically able to perform the work and use the respiratory protective equipment. A physician is to determine what health and physical conditions are pertinent. The medical status of each respirator user is to be reviewed at least annually.

4. The licensee is to use equipment approved under appropriate Approval Schedules in 30 CFR Part 11 of the U. S. Bureau of Mines-National Institute for Occupational Safety and Health and as set forth in 10 CFR 20 Appendix A.

5. Where no equipment of a particular type has been approved under the schedules in 30 CFR Part 11, or where there is no existing schedule for approval of certain equipment, such equipment is not to be used except as authorized by the Commission.

6. Unless otherwise authorized by the Commission, the licensee will not assign protection factors in excess of those specified in 10 CFR 20 Appendix A when selecting and using respiratory protective equipment.

4.1.7 Radioactive Releases

4.1.7.1 Air

Airborne radioactivity discharged to the atmosphere shall be controlled to have an annual average activity at the property boundary as far below the Maximum Permissible Concentration (MPC) for unrestricted areas (as defined in 10 CFR Part 20) as practicable. Continual evaluation of air sampling results shall be used to assess the cumulative amount discharged.

Where calculations indicate SNM can be emitted to the site boundary at concentration levels, averaged over a calendar quarter, which are equal to or greater than at least 10% of the appropriate concentration listed in 10 CFR 20, Appendix B, Table II, effluents shall be continuously sampled. Air effluent shall be monitored after filtration during release.

Anytime the air sampling data indicates that a level of 25% of the MPC listed in 10 CFR 20, Appendix B, Table II, has been exceeded on a quarterly basis, an investigation will be conducted and corrective action will be taken as required.

4.1.7.2 <u>Liquid</u>

Liquid effluent shall be controlled by storage until determined safe for: 1) disposal by solar evaporation or 2) disposal through an approved waste disposal agency, or 3) released to the sanitary sewerage system within the limits specified in 10 CFR 20.

Radioactive discharge to the environment shall be as far below the MPC for unrestricted areas (as defined in 10 CFR Part 20) as reasonably achievable and shall not cause any unrestricted area effluent to exceed 100% of MPC. Continual evaluation of water sampling results shall be used to assess the concentrations and cumulative amount discharged. Any time the data indicate that a level of 100% of the MPC may be exceeded on a quarterly basis, an investigation shall be made to determine the corrective action needed.

4.2 EQUIPMENT

4.2.1 Radiation Detection and Survey

Instrumentation for detection and measurement of radiation shall be provided. Instrumentation other than 4.2.1.4 CWAS systems and microR meters shall be calibrated after repair and routinely at least twice annually by use of a source with calibration data traceable to the National Bureau of Standards. MicroR meters shall be calibrated annually.

4.2.1.1 Dose Measurement

Film dosimeters or thermoluminescent dosimeter (TLD) shall be used to measure personnel occupational external radiation dose. Table II 4.2-1 gives a summary of the typical characteristics of such devices. Self-reading pocket ionization chambers (dosimeters) may be used in addition to film dosimeters or TLDs as determined by licensee to measure X-ray, beta, gamma and neutron radiation.

TABLE II 4.2-1

FILM AND TLD DOSE RANGES

Radiation Type	Dose Range From	To	Energy
Film Dosimeter			
X-ray and gamma	10 mRems	40 Rems	5 KeV to 20 KeV
	10 mRems	40 Rems	20 KeV - 100 KeV
	10 mRems	750 Rems	100 KeV - 3 MeV
Beta	50 mRems	750 Rems	Above 1 MeV
TLD Dosimeter			
X-rays and gamma	10 mRems	10^5 Rems	10 KeV and up
Beta	50 mRems	10^5 Rems	1 MeV and up
Neutron Film/TLD's			
Neutrons	30 mRems	1000 Rems	0.025 eV and up

4.2.1.2 Dose Rate Measurement

Portable survey meters shall be used to measure dose rate. Effective meter ranges are shown in Table II 4.2-2.

TABLE II 4.2-2

PORTABLE METER DOSE RATE RANGES

Radiation Type	Dose Rate Range From	Dose Rate Range To	Energy
X-ray and gamma	0.2 mR/hr	5000 R/hr	6 KeV to 3 MeV
Beta	0.2 mRad/hr	50 Rads/hr	N/A
Neutrons	0.1 mRem/hr	2000 mRems/hr	0.025 eV to 15 Mev

4.2.1.3 Radioactive Material Detection and Assay

Portable instrumentation shall be available and utilized for detection and assay. This instrumentation includes:

X ray and gamma - Geiger-Mueller (GM) counters, scintillation detectors, scintillation detectors coupled with a portable scaler for assay of low levels of radioactivity, scintillation detector coupled with a single-channel analyzer, and ion chambers.

Beta - GM counters and ion chambers.

Alpha - Air proportional and scintillation detectors.

Neutron - mRem or Rem reading neutron monitor.

Fixed instrumentation shall be available and utilized as required for radiological safety purposes. Such instruments shall include counting equipment, spectrometers for analysis of samples, calibration sources(s) to calibrate instruments, and meteorological equipment for measurement of certain meteorological conditions.

4.2.1.4 Criticality Monitoring and Alarm System

The licensee shall maintain in each area where SNM is handled, used, or stored a monitoring system using gamma or neutron-sensitive radiation detectors which will energize clearly audible alarm signals if accidental criticality occurs. This system shall meet the performance requirements of 10 CFR 70.24(a). Each system shall be tested monthly using internal check sources or portable sources. Individual channels of the system will be recalibrated when whichever occurs first:

1. detector and amplifier response to a check source is $> \pm 50\%$ from expected value.

2. detector and amplifier response to internal "keep alive" source or known external radiation field is not within ±100% of expected value.

3. prior to reentry into service following any required maintenance.

The alarm trip levels will be set at levels between 5 mR/hr and 20 mR/hr for non-coincidence systems or they will be set to detect a criticality event specified in 70.24(a)(1) if the system is a coincidence type. The trip levels will be readjusted after each monthly test of the criticality alarm system if the alarm point fails to activate within approximately five (5) seconds more than once out of four trials.

Any area which meets the criteria of 1, 2 or 3 below and which has prior CRSC approval is exempt from the monitoring system described above.

1. General Spaces

   a. The area shall be a defined nuclearly isolated space, area, laboratory, building or facility established to control SNM activities.

   b. The area shall have established a material balance which shall be limited to not more than 700 grams contained U-235, 500 grams U-233, 400 grams plutonium or 400 grams of a combination thereof.

   c. The area's activity shall be under an approved work authorization.

   d. Administrative and procedural and/or physical constraints exist to preclude the introduction of more than the authorized amount of SNM to the controlled area.

2. Hot Cells

   a. Heavily shielded cells designed to contain highly irradiated fuel, reactor components, or byproduct material sources.

   b. The operating and service galleries of the cells shall be alarmed.

3. Other

   a. Any area used for temporary storage of SNM in authorized shipping containers on transport vehicle pending its shipment or delivery to on-site storage area.

In addition, the licensee is exempt from the 10 CFR 70.24 monitoring system requirements when performing system repair or modifications provided that:

1. Special attention is given to minimizing the period of inoperability.

2. No material handling will be allowed in the area while the alarms are inoperative unless such handling is to mitigate a significant Health Safety problem or provide required physical protection and receives prior approval of operating management, the managers of Nuclear Safety & Health Physics, and the Chairman of the CRSC.

3. Facility operating staff will be informed of the special operating circumstances limiting SNM handling.

The licensee also is exempt from the 10 CFR 70.24 alarm audible requirement when planned operations will result in radiation levels above the 5-20 mR/hr monitoring system trip levels provided that:

1. The radiation level is continuously measured and is under observation during the interval of bypass of the alarm's audio;

2. Any unrelated SNM handling in the area shall be suspended during the period of bypassed alarm audio;

3. The bypassing of the audio alarm is accomplished by or under direction of Health Physics and

4. The system will be tested for complete operability at the time it is returned to normal service.

Notwithstanding other statements made in this section, no material handling shall be allowed in any area in which the required criticality alarm system is inoperative.

4.2.2 Air Sampling and Filtration

Air sampling equipment shall be maintained and used where there is hazard of particulate airborne radioactivity. This equipment shall, as a minimum, consist of a sampling head, filter, air volume flow measuring device and pump. The standard method of evaluating the air samples shall be based on removal of the sample filter to suitably calibrated laboratory counting equipment.

Whenever High Efficiency Particulate Air (HEPA) filter systems are used, they shall have the following characteristics:

1. Rated operational efficiency of 99.95% for particles of 0.3 micron size.

2. Fire Resistance

   a. Fire resistant - these filters will be made of fire resistant material capable of withstanding the UL spot test, and capable of continuous operation at 150°F, or

   b. Fireproof - these filters shall be made of fireproof materials (i.e., metal frame with asbestos, ceramic, etc.) and capable of operation at temperatures above 150°F.

3. Differential pressure indicators or monitors

HEPA filters using pressure indicators or monitors shall be replaced when the pressure drop across the filter reaches 4 inches of water. HEPA filters, when used in sequence on vent systems employing soot filters, shall be replaced when the pressure drop across the filter reaches 6 inches of water.

Fume scrubber systems may be one of the following types as necessary to accomplish the purpose:

1. Fume scrubber - wet type

2. Packed tower - wet or dry

3. Mechanical separators

The need for an alarm system shall be evaluated for each application. When used, alarm systems shall provide a warning of equipment malfunction. Sensor and detector systems, when used, shall be capable of detecting abnormal conditions such as excessive heat rise, loss of air flow, blocked filters, and abnormal water levels.

4.2.3 Hoods, Glove Boxes and Containers

4.2.3.1 Hoods

Hoods shall be used to provide controlled ventilation and to preclude contamination of personnel and the surrounding areas. They are typically fabricated of metal, fiberglass, plastic or wood. Each shall typically have a movable window that can be closed. Each hood shall be connected by a duct to a suitable ventilation system.

4.2.3.2 Glove Boxes

Glove box design shall be determined by the particular operation for which containment is required. Glove ports shall be used to provide gloved access to the inside area. The glove box atmosphere shall be maintained at a negative pressure differential to assure containment regardless of leak tightness.

4.2.3.3 Inert Gas Boxes

Inert gas box design shall be determined by the particular operation for which containment is required. Such boxes shall have the capability to be continuously purged or evacuated and injected with an inert gas. The gas must be held in the box at, or slightly above, atmospheric pressure without substantial loss for the duration of the operation contemplated. Pressure relief devices shall be provided which limit positive pressure to plus four inches of H_2O P relative to atmosphere. The pressure relief shall be ducted to an exhaust ventilation system. They may be equipped with gloves for gloved access to the inside area.

4.2.3.4 Frequency of Air Flow Measurements

Surveys of air velocity through hoods or similar enclosures shall be made at least on a quarterly basis. Adequacy of the hood velocity shall be further confirmed by placement of air samples outside of hoods or enclosures. The sampler placement shall cause a conservative measure of the workers breathing zone.

A ventilation survey shall not be completed on hoods/enclosures which are not being used. The hood must first be cleaned to minimize potential for airborne release of radioactive material. In fuel fabrication areas, all openings of ventilation ducts and enclosures which are not completely sealed shall be maintained with a minimum airflow of 25 LFM. The licensee shall determine, on at least a semiannual basis, that this minimum airflow is maintained.

4.2.3.5 Containers

The containers used for the transfer and storage of SNM shall be fabricated of durable material (i.e., metal, plastic, etc.), and provide containment capable of minimizing accidental spillage or dispersal of contents. The containers shall not be opened in any locations other than those designated for the purpose of processing, handling, or storing SNM.

4.3 FACILITIES

4.3.1 General

Present buildings containing radioactive material were constructed in accordance with uniform building code requirements and the fire resistant standards. In cases of new construction, including interior finish, partitions, equipment and mounting shall be fire resistant. All applicable building codes will be complied with, except where more restrictive practice is deemed necessary to assure radiological safety. Building services include provisions for ventilation and air conditioning, fire detection and/or extinguishers and exhaust filtration and waste disposal if required.

4.3.2 Hot Cell Facility

The Hot Cell facility shall contain cells capable of handling up to 1 x 10^6 curies of 1 MeV gamma. Cells shall operate under negative pressure and air shall be exhausted to a vent system which contains prefilters and High Efficiency Particulate Air (HEPA) filters rated at an efficiency of at least 99.95% for particles of 0.3-micron size. Cells shall be constructed of steel and concrete and shall use shielding windows to afford radiation protection. Devices shall be located throughout the facility and in the exhaust system to monitor radiation and system operation.

4.3.3 Storage Areas

Storage areas shall be provided to accommodate any physical or chemical form of SNM authorized under this license. All storage facilities shall have limits established within these license specifications and applicable governmental regulations.

4.3.4 Contamination Limits for Release to Unrestricted Use

The releases of facilities and equipment for Unrestricted Use from the plant site or to Unrestricted Areas onsite shall be in accordance with Annex C, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," dated July 1982.

5. NUCLEAR SAFETY - TECHNICAL REQUIREMENTS

The continued nuclear safety of the licensee's operations shall be assured by limits and procedures documented in accordance with the specifications contained in the following sections:

5.1 DEFINITIONS

The following definitions are used herein for purposes of this license:

Array - Two or more interacting fissile material accumulations.

Criticality Area - Any physically identified area or location involving SNM in quantities exceeding 500 gm of U-235, 300 gm of U-233, or 300 gm of plutonium or a prorated combination of such materials under the direction of an Operation Manager. One individual is designated to be responsible for custody and control of the SNM in each criticality area.

Criticality Control - The administrative and technical requirements established to minimize the possibility of achieving inadvertent criticality in the environment analyzed.

Normally Subcritical Values - Those minimum values which provide safety under normal conditions of operation.

Operating Limit - A ceiling imposed by operating procedures to limit the quantity (usually mass or volume) of SNM within a given storage location.

Storage Limit - A ceiling imposed by storage procedures to limit the quantity (usually mass or volume) of SNM within a given storage location.

Safety Factor - The ratio of the critical value of a quantity to the maximum allowed value; i.e., it is the inverse of the maximum allowable fraction of the critical value of that quantity.

Safety Margin - The difference between the critical value and the maximum allowed value for a quantity.

5.2 BASIC ASSUMPTIONS

The basic assumptions that shall be utilized in arriving at particular criticality limits are as follows:

Accident - At least two unlikely, accidental, concurrent, and independent events must occur before a criticality hazard could exist.

Critically Safe Dimensions (For U-235 systems only) - Containers and other vessels are critically safe if volume is 1 gallon (3.8 liters) or less, or, diameter is 5.0 inches (12.7 cm) or less, or, for slabs the thickness is 1.5 inches (3.8 cm) or less.

Mass - Double batching must be assumed unless it is not physically credible.

Ever Safe Mass - A total mass of 350 g of U-235 at enrichments up to 93% is safe under all conditions. A total mass of 457 g of U-235 at enrichments up to 20% is also safe under all conditions. For enrichments other than 93% or 20%, the ever-safe mass should be redefined per Fig. 22 of

LA-10860.

Geometry - The most reactive credible geometry must be assumed. In an unrestricted situation this is a sphere.

Volume/Distribution - Full dispersal and/or possible agglomeration must be considered.

Moderation - The most reactive credible degree of water and/or other moderating materials such as carbon must be considered.

Enrichment - Fully enriched SNM (93.5% U-235 or 97% Pu-239 or 100% U-233) is assumed unless it can be shown that a lower enrichment value cannot be exceeded. Enrichment above 93.5% U-235 shall be assumed to be 100%.

Reflection - Full water reflection in addition to the reflection properties of containment material associated with the unit must be used unless water reflection is not credible. Where credible, reflection by carbon or concrete must be considered.

Interaction - Interaction between non-isolated fuel regions must be considered.

Poisons - The presence of neutron-absorbing poisons shall not be considered if their absence is credible.

Plutonium - From a criticality standpoint, all isotopic concentrations of plutonium shall be handled under limits that assume 97% Pu-239.

5.3 REQUIREMENTS FOR WASTE AND SCRAP

Special nuclear material being accumulated for waste disposal or scrap recovery shall be accrued on an ever-safe mass limit or ever-safe dimension limit basis. Before any individual accumulation is added to a storage array that is limited by any criteria other than ever-safe mass or dimension, the SNM content shall be verified by analysis methods maintained under statistical control assuring the following conditions: (1) SNM content is verified within known limits of error, and (2) the errors of the measurement are used in setting station limits, in order to assure that the safe limits cannot be exceeded under the worst conditions of measurement error.

5.4 LIMITING CRITICAL QUANTITIES

Accumulations and arrays of SNM shall be maintained in a critically safe condition by conformance to one or more of the specifications in items 2 through 6 which follow.

After a proposed operation is studied and the most reactive credible situation determined, the quantity to be limited is selected and its critical value determined. The safety factor is then applied to the most reactive credible situation to obtain the maximum allowable safe limit. Alternatively, the nuclear criticality safety of the proposed operation is assured by showing that in the most reactive credible situation the effective multiplication factor is equal to or less than the limit defined in Section 5.4.4(a). In this case, the geometries and masses used in the nuclear safety evaluation define the maximum allowable limits of the operation. The most reactive credible situation takes into account normal and abnormal conditions and the criteria for determining such conditions are given in part 6 below.

1. Reference documents on bench mark critical experiments, collections of experimental data, and analyses may be used providing they are reviewed for adequacy. Factors to be considered are:

   a. Consistency with other data.

   b. Freedom from anomalies.

   c. Applicability to the case of interest.

   d. Accuracy or error margin.

   e. Analytic methods shall have been demonstrated to be valid either in the document in question or in a prior body of data.

2. Critically safe limits based on mass may rely on form (e.g., solid metal) and shall satisfy one of the following criteria:

   a. If double batching is credible, the mass limit shall have a safety factor of 2.3 below a critical mass.

   b. If double or additional partial batching is not credible, the mass limit with allowances for measured uncertainties shall have a safety factor of 1.09 below (92% of) a critical mass. Provision shall be made for the largest batch size possible in the container geometry at the station.

   c. The minimum critical mass, in grams, obtainable with optimum water moderation and reflection, as demonstrated for water mixtures in Fig. 8 of TID-7028)[1] and the applicable safety factor limits are given in Table II 5.4-1.

TABLE II 5.4-1

MINIMUM CRITICAL MASS OBTAINABLE WITH OPTIMUM WATER MODERATION AND REFLECTION (Mass Given in Grams)

Isotope	Critical	2.3 Safety Factor	1.09 Safety Factor
U-235 (93.2%)	805	350	740

d. Uranium of enrichment greater than 93.5% shall be handled under reduced mass limits which assume 100% U-235. The factor to be used is:

$$\text{Mass} = \text{Mass Limit}_{93.5} \times \frac{1}{1.089}$$

e. Uranium of enrichment less than 93.5% may have increased safety limits provided it can be demonstrated that presence of uranium above a given low enrichment is not credible. Such limits shall be determined using either documents referenced per Section 5.4.1 or an approved calculation technique meeting the criteria of Section 5.4.4.

f. If double or additional partial batching is not credible, the Th/U atomic ratio is ≥ 3.6 and the H/U atomic ratio is ≤ 20, then the maximum safe batch of HTGR type particles is 790 g of U-235 at 93.2%

[1]Paxton, H. C., et al., "Critical Dimensions of Systems Containing U-235, Pu-239, and U-233," USAEC Report TID-7028, Los Alamos Scientific Laboratory and Oak Ridge National Laboratory, June 1964.

enrichment.

3. Critically safe limits based on parameters other than mass shall satisfy one of the following criteria on the maximum allowable fraction of the critical value (to be applied after calculational imprecisions in geometrical dimensions, uncertainties in the experimental data, and possible credible accidental changes are taken into account):

   a. Volume - safe limit shall be limited to 75% of critical volume.

   b. Diameter - safe limit shall be limited to 93% of critical diameter.

   c. Thickness - safe limit shall be limited to 88% of critical thickness.

   d. The geometrical safe limits shall apply to units meeting the following criteria:

      i. The units shall be of rigid construction.

      ii. The unit shall be evaluated for all possible credible accident conditions to assure that the safety margin (difference between critical and safe limits) is not exceeded.

      iii. When the sum of possible calculational imprecisions, imprecision in geometrical dimensions, and possible credible accidental changes exceeds the safety margin, then additional safety margin must be included. Specifically, slabs with critical dimensions less than 1.5 in. shall always include added safety margin for imprecisions in geometrical dimensions.

4. Critically safe limits based on reactivity shall be determined by analysis considering optimum moderation and reflection, including possible effects of water. If the optimum is not possible, the analysis shall consider the most reactive case of moderation and reflection credible and shall include documented justification for the assumption. The analysis shall be made with a method which satisfies one of the following criteria:

   a. Systems of SNM shall be analyzed by computer codes that have been validated by application to several experiments or benchmark cases of similar geometrical configuration and material content; the maximum negative bias, defined as $k_{calc} - k_{exp}$, determined in the validation will be applied to the analysis.

      The reactivity of the analyzed system shall satisfy

      $$k_{calc} + \Delta k_{uncertainty} - \Delta k_{bias} \leq 0.95$$

      In this relationship, $k_{uncertainty}$ is the statistical uncertainty (2σ) in the calculated value, and k_{bias} is the maximum negative bias determined by the validation of the code. Positive biases, i.e., overstatements of the experimental reactivities, shall not be corrected for. Uncertainties in reference data and validations shall also be applied.

   b. All geometries analyzed by hand calculations shall have a k_{eff} of 0.90 or less.

   c. Moderation control shall be used only when the station is fully enclosed, has no piped source of liquids, has suitably trapped ventilation or exhaust systems, and controls exist on the introduction of moderating materials by the operator.

5. SNM in shipping containers specifically licensed, generally licensed, or exempt may be received and stored provided the entire array involved is not more reactive than that permitted by the shipping authorization, or unless storage is justified by separate nuclear analysis meeting other requirements of Section 5.4.

6. The criteria for determining the most reactive credible geometry are as follows:

   a. Normal and abnormal conditions (including accident and operating errors) are taken into account.

   b. All possibly postulated accident conditions are evaluated with respect to their credibility for occurrence and those deemed credible are taken into account.

   c. The most reactive geometry is determined for the parametric space defined by considering all possible credible values for these variables: mass, volume, density, geometry, physical form, reflection, moderation, chemical and isotopic composition, concentration, interaction, and poisons.

   d. In addition to all possible normal operating conditions, the most reactive geometry is determined considering credible abnormal conditions possible within the scope of the nature of the processing area, the operating procedures, equipment for handling material, and training and supervision of operating personnel.

7. The criteria for determining that double batching is not credible is as follows:

   a. Double batching is not credible when it takes two or more unlikely events to occur before double batching can occur.

   b. Double batching is not credible when geometry constraints preclude placing more than one safe mass in a fixture or unit controlled by safe mass limits. If the constraints are such that a partial second batch may be added, such overbatching must be considered in the nuclear safety analysis.

   c. Double batching is not credible if an inherent characteristic of the operation performed makes the addition of a second batch or mass very difficult or impossible.

5.5 ISOLATION AND INTERACTION

Nuclear isolation of accumulations or arrays of SNM is determined as follows:

1. A process unit or a process station composed of several interacting units shall be considered to be nuclearly isolated from other units, stations, and SNM arrays if separated therefrom by one of the following or its equivalent:

   a. Twelve inches of water.

   b. Concrete having a density of 140 lb/ft^3 or more and a thickness of sixteen inches or more. Where the concrete does not meet this criteria the stations shall be considered to be interacting and the safety must be justified through specific analysis or through other validated methods, such as equivalent solid angles of interaction.

c. An edge-to-edge separation that is the larger of:

i. Twelve feet, or

ii. The greatest distance across an orthographic projection of either accumulation or array on a plan perpendicular to a line joining their centers.

2. SNM in a plane storage array shall be considered to be nuclearly isolated from similar storage arrays if the arrays are separated by concrete having a density of 140 lb/ft^3 or more and a thickness of:

a. Sixteen inches or more, or

b. Eight inches or more and either array is separated from the concrete isolation wall by a distance greater than the array's minor lateral dimension excluding the smallest dimension of the array, i.e., its thickness, or

c. If the conditions of a or b are not satisfied, concrete walls may be used to separate SNM storage arrays if the reactivity of the interacting arrays is justified through analysis per 5.4, item 4 which includes the nuclear effect of the separating concrete.

3. Accumulations or arrays of SNM that do not meet the isolation limits of (1) or (2) above shall be considered as interacting and shall satisfy one of the following:

a. Use of the limits in size, weight and moderation of each unit specified in Table II 5.5-1 in a single plane array on 16 inch horizontal and 18 inch vertical, or greater, center-to-center spacing and 8 inch., or greater, surface-to-surface spacing.

TABLE II 5.5-1

MASS AND CONTAINER VOLUME LIMITS FOR INDIVIDUAL UNITS

Degree of Moderation (H/X)				
More Than	Not More Than	U-235	Pu	U-233
		Mass Limits (kilograms)		
-	2	10.0	2.5	2.8
2	3	9.0	2.5	2.5
3	5	7.3	2.5	2.2
5	8	5.2	2.5	1.8
8	15	3.6	2.4	1.3
		Volume Limits (liters)		
15	-	3.8*	2.4	1.3

*3.6 is used for spherical geometries.

b. Reactivity analysis per 5.4, item 4, or when the reactivity of the individual units which interact is known, the effect of the interaction on the reactivity of the individual units may be evaluated by analysis of the fractional reduction in leakage from each unit caused by the multiplication and location of the other unit.

c. Solid angle analysis with the following criteria:

	Max. Solid Angle (steradians)
k_{eff} <0.3	$\Omega \leq 6.0$
0.3 $\leq k_{eff} \leq$0.8	$\Omega \leq (9 - 10\ k_{eff})$
k_{eff} >0.8	Must be analyzed per item (a) above.

i. The reactivity for interaction purposes shall be determined assuming the most reactive degree of moderation within the container that could result from accident, human error, or leakage of water into the container.

ii. The units to be interacted shall be separated by 12 inches or more surface-to-surface unless so arranged and protected that water or hydrogenous moderator cannot intrude between the containers, in which case 8 inches shall be permissible.

iii. The solid angle analysis shall be based on the total solid angle subtended at the center of one unit by the adjacent units and shall include all units containing SNM that are within a distance of 12 feet or which contribute 0.005 steradian or more.

iv. Solid angle analysis shall be reviewed and approved by an independent party.

v. Each unit in the interacting system shall be separately justified as being safe when fully reflected by water.

vi. Unit k_{eff} shall be those for the isolated bare unit, including the unit container.

vii. The calculated solid angle may be determined by the approximation formulae in Fig. II 5.5-1.

viii. "Shadowing" of units in an array where a reflector is in the vicinity shall not be allowed.

d. Nuclear criticality safety limits may be expressed in terms of an allowed surface density, , of fissile material by the relation

$$= 0.54\ \sigma_0(1 - 1.37f) \qquad \text{I}$$

and the corresponding center spacing of units, d, in mm, from

$$d = \left(\frac{nm}{\sigma}\right)^{1/2}$$

by

$$d = 1.37\ \left(\frac{nm}{\sigma_0(1 - 1.37f)}\right)^{1/2} \qquad \text{II}$$

where

σ_0 = the surface density of the water-reflected infinite slab in g/cm^2

f = the ratio of the mass of a unit in the array and the critical mass of an unreflected sphere of the same material,

n = the number of units in the direction of the projection onto a wall or the floor of the storage area, and

m = the mass of a unit in g.

Equations I and II are applicable to infinite planar arrays reflected by water at least 155 mm thick or its nuclear equivalent. The reflector is located no closer to units in the array than the boundaries of the cells associated with the units.

5.6 EQUIPMENT INTEGRITY

Factors affecting equipment integrity, as elaborated below, shall be considered by each reviewer and approver. Where needed, special expertise snall be obtained through normal company channels and functions specialized in the areas of concern.

Structures, equipment, and containers that directly affect criticality control shall be designed with an adequate safety margin to assure against failure under credible accident conditions and loads. The factors to be considered shall include fire, melting, impact, seismic loading, corrosion, and the effects of liquids, solutions, and precipitates. Seismic design shall be adequate to assure criticality control during and after an earthquake of Mercalli magnitude VIII.

Materials of construction shall be selected to provide integrity under credible accident conditions.

Neutron absorbers used to satisfy 5.4, item (4), shall be of such a nature and design as to satisfy the following requirements:

a. Soluble poisons shall not be used.

b. If borosilicate glass Raschig Rings are used, the initial specifications and environments shall satisfy the chemical and physical requirements specified in ANSI/ANS-8.5-1979.

c. The design shall assure against loss of integrity under credible accident conditions.

d. The design shall also provide for periodic visual or remote inspections of the poison and equipment.

e. All poisons shall meet prescribed initial specifications and acceptance tests.

f. All poisons shall be periodically inspected to assure against settling, solid accumulation, and physical or chemical property changes. Records of these inspections shall be maintained and the frequency shall not exceed 13 months.

g. If the poisons do not come into direct contact with SNM or chemically active solutions, visual inspection is adequate.

5.7 ANALYSIS, DETERMINATION, VALIDATION AND VERIFICATION

The operating departments are responsible for the nuclear safety of their operations and shall be responsible for definition, analysis and documentation of the detailed nuclear safety analysis of any activity initiated under this license per Section 3.

Nuclear Safety Manager shall review, provide, or obtain separate verification of the analysis. The Criticality and Radiation Safeguards Committee (CRSC) shall verify as a second independent party or obtain separate verification of the analysis. Upon documentation of satisfactory two level review, the Nuclear Safety Manager shall approve the Work Authorization.

1. <u>Validation of Calculated Methods</u> - Designated organizations shall maintain suitable analytical codes, nuclear cross-section data, and records of correlation of these analytical methods with suitable direct experimental data. This shall be done in accordance with documented company policies and procedures which satisfy Reg. Guide 3.41 and ANSI Standard N 16.9 - 1975.

   a. All calculational methods shall have demonstrated adequate agreement with experimental results prior to their approval for use in making detailed criticality calculations. Any potential non-conservatism shall be determined at the 2 level and provided for per section II 5.4.4.

   b. All computer calculational methods shall be approved by the licensee's competent organization responsible for nuclear codes prior to their use in making criticality calculations. Approval of computer calculational methods is based upon meeting the following criteria.

      i. The methods are well proven to provide adequate, reliable results in criticality safety analyses.

      ii. Nuclear cross sections used in criticality calculations shall be from approved nuclear data files.

c. Hand calculational methods may be used to extend published data to:

   i. Similar situations with slightly different geometries by making the appropriate geometric buckling correction, or

   ii. To obtain conservative estimates of k_{eff} for simple systems using the Fermi age-diffusion method.

5.8 OTHER SPECIAL NUCLEAR SAFETY REQUIREMENTS

1. Those situations involving large, dilute carbon systems even when the U-235 mass density is less than 0.02 kg/liter, shall be analyzed for the effects of the carbon.

2. Large, very dilute systems involving Be, BeO, or D_2O are restricted from the present operation.

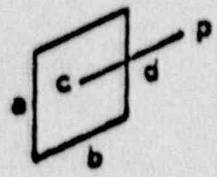
3. Standardized limits with documented CRSC approval may be established and such limits may be applied in any area without the need for additional detailed criticality calculations provided that the values for k_{eff} (both bare and reflected) and the allowable interaction values are adhered to. The application of a standard limit shall be accomplished by utilizing the documentation and approval procedures set forth in these specifications.

4. All SNM stored under mass limits with an associated moderation limit shall be stored in watertight metal containers.

5. Each defined area, in which more than 350 gms of U-235, 250 gms of U-233, 220 gms of Pu, or 220 gms of combined SNM is used or stored, or alternatively, each process station in which the special nuclear materials are limited by procedural controls shall be suitable posted with applicable operating limits. This posting may be more conservative than the license limits and may be stated in terms that are characteristic of the materials used in the area.

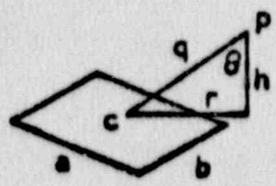
6. Each container for fissile material shall bear a label, which by its characteristics indicates that the contents are fissile. The quantity of the contents is to be shown or a unique identification shall be affixed so that the contents are readily traceable in the records.

TYPE I GENERAL



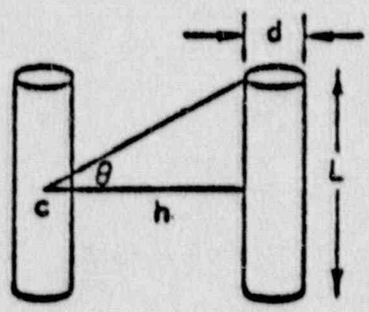
$$\Omega = \frac{ab}{d^2} = \frac{\text{cross-sectional area}}{(\text{separation distance})^2}$$

TYPE II PLANES



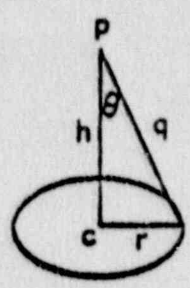
$$\Omega = \frac{ab}{q^2} \cos \theta$$

TYPE III. CYLINDER



$$\Omega = \frac{2d}{h} \sin \theta$$

TYPE IV DISC OR SPHERE



$$\Omega = 2\pi\,(1 - \cos \theta)$$

Fig. II 5.5-1 Solid Angle (Steradians) Calculations

6. ENVIRONMENTAL PROTECTION

An environmental surveillance program shall be maintained to evaluate the effectiveness of the radiological safety program and to provide information to assist in timely corrective action in the event of accidental releases.

The program shall include (1) air sampling, (2) water (including sewage) sampling, and (3) sampling of soil, vegetation, water, and (4) external gamma radiation. Table II 6.0-1 lists the minimum detection sensitivity applicable to the above program samples.

The licensee shall prepare and submit to the Chief, Fuel Cycle Safety Branch, Division of Industrial and Medical Nuclear Safety, NMSS, U. S. Nuclear Regulatory Commissin, Washington, DC 20555, a report whenever there is any new residential development within one mile around the site that constitutes a significant change in parameters that may be affected by the release of radioactive materials into the environment.

6.1 AIR SAMPLING

Air shall be sampled at no less than 15 locations on, adjacent to, and near the site. The sampler filters shall be changed weekly. The filters shall be analyzed for long-lived alpha and beta radioactivity and shall be composited and gamma scanned monthly. The analysis' minimum detectable concentration for weekly samples shall be 0.025 pCi/m^3 or less.

6.2 WATER SAMPLING

Effluent water shall be sampled daily during the normal workweek. The gross alpha and gross beta concentrations shall be determined.

Where radioactive materials are authorized for release to the municipal sewerage system, such materials will be held in a tank until samples have been taken and analyses performed to assure their release is in compliance with applicable laws and regulations.

TABLE II 6.1-1

MINIMUM DETECTION SENSITIVITY

Sample Type	Radiation	Sensitivity
Water[1]	alpha	6 pCi/liter
	beta	4 pCi/liter
	gamma	3 pCi/liter
Soil	alpha	10 pCi/g
	beta	2 pCi/g
	gamma	0.1 pCi/g
Vegetation[2]	alpha	15 pCi/g
	beta	2 pCi/g
	gamma	0.5 pCi/g
Air[3]	alpha	0.025 pCi/M^3
	beta	0.010 pCi/M^3
Radiation	gamma	10 mR
	beta	40 mR

[1] Dependent on total solids content.

[2] May vary dependent upon accompanying organic residue.

[3] Seven day sample.

6.3 SOIL, VEGETATION AND WATER

An annual survey shall be made of samples of typical soil, vegetation, water, and their radiation levels. Gross-alpha, gross-beta, and gamma spectral analyses shall be routinely performed on each sample. Gamma radiation shall be measured at each station. There shall be not less than 16 stations used in the annual environmental survey.

A quarterly survey shall be conducted if the air sample results, from stations at the site boundary, show quarterly average releases equal to or greater than 25% of the unrestricted area MPC identified in 10 CFR 20, Appendix B.

6.4 EXTERNAL GAMMA

Film or thermoluminescent dosimeters shall be used to gain information on the environmental integrated dose at selected air sampling locations. The dosimeters are changed quarterly.

6.5 LOCATION CRITERIA FOR ENVIRONMENTAL SAMPLING STATIONS

1. Each sampling station shall be established with consideration for:

   a. Type and quantity of material which may be discharged to the environment.

   b. Postulated mode of release.

   c. Characteristics of the local environment, including local population distribution and land use.

   d. Ability of the station to detect the release of radioactive material from the licensed facility.

2. Sampling stations which show statistically significant results above natural background radiation levels will not be deleted (unless specific approval is obtained from the NRC). Any site deletion will be documented.

3. Where construction or other activities require removal of a sampling station, a new station with a new identification shall be established as close as practicable to the old location.

4. New sampling stations may be added where meteorology, terrain, and population buildup indicate their possible necessity. These new locations may be deleted after 2 years of operation if the sampling results satisfy criteria 2 above.

6.6 NONRADIOLOGICAL MONITORING

6.6.1 Hydrogen Chloride in Ambient Air

Hydrogen Chloride concentration shall be monitored at least quarterly at two stations on the plant boundary in the vicinity of the SVA Facility (HTGR Fuel Manufacturing) if the HCl furnace is operating.

The sampling apparatus at each hydrogen chloride collection station shall consist of:

1. A Greenberg-Smith Standard Impinger
2. An Air Sampling Pump
3. An Integrating Dry Gas Meter

Hydrogen chloride in ambient air shall be collected by passing approximately 4 m^3 of air through 100 ml of 0.01 N NaOH absorption solution in the standard impinger at a rate of approximately 30 liters/min. The determination of HCl concentration shall be by chloride titration or an equivalent method which has a sensitivity of detecting 0.01 ppm HCl in air.

6.6.2 Liquid Waste Monitoring

GA shall monitor at least semiannually its industrial wastes with regard to pH, suspended solids, biochemical demand and ether solubles to show compliance with the City of San Diego's Industrial Waste Discharge Permit.

7. OTHER SPECIAL COMMITMENTS

7.1 PACKAGING AND SHIPPING

Packaging and shipping of SNM will be accomplished within the requirements of 10 CFR Parts 71 and 73; 49 CFR Parts 171 through 400; and other applicable Federal regulations.

The licensee shall assure compliance with the packaging, labeling, shipping, protection, and documentation requirements for radioactive material shipments.

The licensee shall develop, revise, implement, and enforce written packaging and transportation procedures.

The licensee shall develop and maintain procedures (including emergency procedures) for off-site transportation of SNM which shall include provisions for radiological safety and physical protection.

7.2 PLUTONIUM

7.2.1 Sealed Source

Plutonium classed as a sealed source means plutonium that is encased in a capsule designed to prevent leakage or escape of the plutonium. Typically such sources have double or triple enclosures. All sealed sources shall be purchased from licensed vendors and the licensee shall not open the enclosing bodies or conduct any operation that will affect the integrity of the sealed source. Such sources shall be subjected to leak testing per Section 4.

7.2.2 Encapsulated Sources

Plutonium classed as encapsulated sources shall be obtained from other qualified fabricators. Such sources shall be within enclosing bodies, properly sealed, in an assembled state. The licensee shall not open the enclosing bodies or conduct any operation which will affect the integrity of the encapsulation except in a Health Physics-approved glove box for the purpose.

Satisfactory encapsulation may include sealed metallic containers, HTGR type coated particles, plated calibration sources and equivalent types of enclosures. The encapsulation designs shall be verified by analysis and/or testing. All encapsulation designs shall be approved in writing by LSNC and reviewed by CRSC prior to use.

The analysis and/or testing shall meet, insofar as practicable, the criteria below:

1. The encapsulation material shall be chemically compatible with the contained material.

2. The encapsulation shall contain the end of life pressure resulting from gaseous radioactive decay or radiolytic decomposition products.

3. The chemical form of the loading, maximum gram loading and the design life of the encapsulation shall be specified.

4. The encapsulation shall withstand heating in air to 1475°F for 30 minutes under conditions of end of life internal pressure or be kept in fireproof containers or glove boxes.

7.2.3 Storage

Plutonium shall not be stored outside, other than in approved shipping containers.

7.2.4 Unencapsulated Plutonium Requirements

A bioassay program for plutonium shall be established, and personnel working with unencapsulated plutonium shall have urine samples analyzed at least quarterly. In the event of a known plutonium release or if a urine sample exceeds 0.2 dpm daily excretion, then the lung burdens of the exposed individuals shall be assessed by whole-body counting.

Personnel, including visitors, shall wear shoe covers and smocks in the plutonium laboratories. In addition, all individuals leaving the plutonium laboratory areas shall be checked for contamination and corrective action taken if plutonium activity is detected.

8. SUPPORTING PLANS

8.1 RADIOLOGICAL CONTINGENCY PLAN

The licensee shall maintain and implement the response measures described in the Radiological Contingency Plan dated June 1988. The licensee shall also maintain implementing procedures for the Radiological Contingency Plan as necessary to implement the Plan. This Radiological Contingency Plan and associated implementing procedures supersede the emergency planning requirements of 10 CFR 70.22 (i) as they refer to on-site planning and notification procedures. The licensee shall make no change in the plan that would decrease the effectiveness of the Plan without prior Commission approval as evidenced by a license amendment. The licensee may make changes to the Plan without prior Commission approval if the changes do not decrease the effectiveness of the Plan. The licensee shall maintain records of changes that are made to the Plan without prior Commission approval for a period of 2 years from the date of the change and shall furnish the Chief, Fuel Cycle Safety Branch, Division of Industrial and Medical Safety, NMSS, U. S. Nuclear Regulatory Commission, Appendix D of 10 CFR Part 20, a report containing the description of each change within 6 months after the change is made.

The licensee's Emergency Plan shall consist of a document containing, but not necessarily limited to, information and procedures needed to demonstrate compliance with the applicable elements prescribed in 10 CFR 50, Appendix E, Section IV, subsections A through H. Paragraphs 2 and 3 of subsection D are deemed inapplicable.

8.2 PLAN FOR CONTROL AND ACCOUNTING OF SNM

The licensee shall, in accordance with 10 CFR 70.22(b) develop and maintain plans for the control and accounting of SNM.

Currently the licensee has NRC-approved Fundamental Nuclear Material Control (FNMC) and measurement control plans as required by 10 CFR 70.58 and 70.57 respectively.

8.2.1 Fundamental Nuclear Material Control Plan (FNMC)

The licensee shall follow Revision A of his "Fundamental Nuclear Material Control Plan," dated October 1982; as revised by the following submittals:

1. Submittal of August 17, 1984 (pages 3-6, 3-7, 3-8b, 3-8c, 3-8d, 3-8f, 4-6b, 4-17, 4-19a, and 4-20 --- all dated "August 1984"),

2. Submittal of December 21, 1984 (pages 4-4a, 4-5, and 4-19 --- all dated "August 1984"),

3. Another January 25, 1985 submittal (pages 7-7 through 7-8 --- all dated "February 1985"),

4. Submittal of December 5, 1985 (pages 1-1, 1-5, and 1-10 --- all dated "December 1985"),

5. Submittal of March 7, 1986 (pages 1-6, 3-2, 3-8a, 3-8e, 4-8, 4-18, 5-1, 5-5 through 5-7, 6-2, 6-4 through 6-6, 7-1, 7-4 through 7-6, and Figures 2.1.1-1, 6.1.1.1, and 6.1.1.2 --- all dated "February 1986"),

6. Submittal May 22, 1986 (pages 2-1, 2-3 through 2-5, 2-7, 4-8, 5-2, 6-1, 6-3, 7-16, and 7-18 --- all dated "May 1986"),

7. Submittals of August 5 and 13, 1986 (pages 3-1, 5-3, and 7-10 through 7-21 --- all dated "August 1986"),

8. Submittal of December 10, 1986 (Figure 1-1 --- dated "November 1986"), and

9. Submittal of April 29, 1988, as amended by letter of September 20, 1988 (pages i, ii, iv through xi, 1-2, 1-3, 1-4, 1-7, 1-8, 1-9, 2-2, 3-1, 3-3, 3-8, 4-1 through 4-4, 4-7, 4-15, 5-4, 6-3, 6-7, 6-8, 7-3, 7-9, 8-1 through 8-4, and Figures 1.1 and 2.1.1-2 --- all dated "April 1988").

This Plan may be further revised pursuant to the provisions of 10 CFR 70.32(c).

Upon reporting an unbiased, adjusted Inventory Difference (ID) value for any regular material balance period that exceeds 1.50 times its applicable regulatory LEID limit (or LEMUF limit) and which also exceeds 300 grams HEU, or U-235 contained in HEU or 9000 grams U-235 contained in LEU, the licensee shall immediately initiate a reinventory of the Plant giving rise to such ID. Unbiased, adjusted IDs shall be obtained by applying any appropriate bias correction to the Inventory Difference value in accordance with Section 4.2.4.1 of the Licensee's FNMC Plan, and adjusting the resulting ID for any prior period adjustment made to the Beginning Inventory value.

8.2.2 MEASUREMENTS

Notwithstanding the requirements of 10 CFR 70.58(e) the licensee need not measure the special nuclear material content of encapsulated SNM sources which are unopened for research and development, calibration activities, or customer service.

Notwithstanding the requirements of 10 CFR 70.51(e)(2) to inventory all SNM possessed by the licensee, waste materials in Plant III destined for ultimate disposal need not be included in physical inventory listings.

For those materials measured by non-destructive assay (NDA), the licensee may calculate the element content based on the U-235 NDA measurement and a prior isotopic abundance measurement that can be demonstrated to be representative of the material being measured by NDA.

8.2.3 MEASUREMENT CONTROL

All identifiable covariance effects shall be taken into account in the calculation of LEID (or LEMUF).

LEID (or LEMUF) calculations shall not be required for material balances associated with Plants III, IV and V provided the associated Inventory Difference value does not exceed 300 grams uranium or U-235.

Notwithstanding the requirements of 10 CFR 70.57(b)(4) to determine systematic sampling errors and perform engineering tests to establish or verify the applicability of existing mixing and sampling procedures, the licensee shall follow Section 4.3 licensee's FNMC Plan.

Notwithstanding the requirements of 10 CFR 70.57(b)(8) to measure standards and replicates for volume systems, to determine separate random errors for sampling and analytical, and to generate random errors for bulk and NDA from replicate measurements of process materials, the license shall follow Sections 4.2.2 and 4.4.1 of the licensee's FNMC Plan.

Notwithstanding the requirements of 10 CFR 70.57(b)(10) to perform bias calculations and corrections and determine limits for systematic errors, the licensee shall follow Section 4.2.4 of the licensee's FNMC Plan.

In satisfying the requirement of 10 CFR 70.57(b)(3) and the commitment contained in Section 4.1.4 of the licensee's FNMC Plan, the licensee may, in lieu of an on-site review, utilize telephone discussions with appropriate contractor personnel and inspection of copies of relevant documents requested by the licensee and provided by the contractor laboratory.

8.2.4 RECORDS AND REPORTS

The licensee shall use his or his designated agent's measured values for SNM accounting, except in Section 8.2.2., i.e., notwithstanding the requirements of 10 CFR 70.58(e) the licensee need not measure the special nuclear material content of encapsulated SNM sources which are unopened for research and development, calibration activities, or customer service.

8.2.5 INTERNAL CONTROL

Notwithstanding the requirements of 10 CFR 70.51(e)(1)(iv) and 70.58(h) to maintain current knowledge of the identify, quantity, and location of all discrete items and containers of

SNM, the licensee shall follow Section 7.3, and all subsections therein, of the licensee's FNMC Plan.

8.2.6 MANAGEMENT

Notwithstanding the requirements of 10 CFR 70.57(b)(2), (b)(3), and 70.58(c)(2) to conduct management reviews and audits at intervals not to exceed twelve months, the license shall follow Sections 1.2.5, 4.1.3, 4.1.4, 8.2.1, and 8.2.1.1 of the licensee's FNMC Plan. For the purpose of this Condition, the maximum 15-month interval between Review Committee briefings is hereby defined as a period not to exceed 460 calendar days.

8.3 PHYSICAL SECURITY PLAN

The licensee has, in accordance with 10 CFR 70.22 (G) and (K), an NRC-approved physical security plan entitled "Fixed Site and Transportation Plan for the Protection of Special Nuclear Material of Low and Moderate Strategic Significance", dated May 1989. This plan is withheld from public disclosure pursuant to 10 CFR 73.21. The licensee shall maintain and fully implement all provisions of this commission approved plan, including changes made pursuant to 10 CFR 70.32(e).

8.4 DECOMMISSIONING PLAN

At the end of plant life, the licensee shall decontaminate the site and facilities, authorized as a place of use for special nuclear material, in accordance with the general decommissioning plan submitted by letter dated July 25, 1986, and its supplement dated October 15, 1986, so that those facilities and grounds can be released for unrestricted use. The corporate commitment that funds will be made available for decommissioning the facility is provided by letter dated September 10, 1986.

The licensee has submitted a Decommissioning Plan (Rev. July 1986) as requested by the Commission. The plan has estimated the costs of decommissioning GA major facilities and shows that such costs are small in relation to the assets of the licensee's owners. Accordingly no bond or cash reserve is maintained. The plan's methodology and costs are based upon a NRC criteria established in "Guidelines for Decontamination of Facilities and Equipment Prior to Release to Unrestricted Use or Termination of Licenses for Byproduct, Source or SNM" dated July 1982.

The release of the facilities and equipment for unrestricted use from the plant site or to unrestricted areas onsite shall be in accordance with the enclosed Annex C, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material", dated July 1982.

9. GLOSSARY OF FREQUENCY DEFINITIONS

Daily means once each day with no period more than 36 hours.

Weekly means once each calendar week with no period greater than nine days.

Monthly means once each calendar month with no period greater than 40 days.

Quarterly means once each calendar quarter with no period greater than 106 days.

Semiannually means twice each calendar year with no period greater than 210 days.

Annually means once each calendar year with no period greater than 15 months.

Annex A

LEAK TESTING SEALED PLUTONIUM SOURCES

November 1979

A. Each plutonium source shall be tested for leakage at intervals not to exceed six (6) months. In the absence of a certificate from a transferor indicating that a test has been made within six (6) months prior to the transfer, the sealed source shall not be put into use until tested.

B. The test shall be capable of detecting the presence of 0.005 microcuries of alpha contamination on the test sample. The test sample shall be taken from the source or from appropriate accessible surfaces of the device in which the sealed source is permanently or semipermanently mounted or stored. Records of leak test results shall be kept in units of microcuries and maintained for inspection by the Commission.

C. If the test reveals the presence of 0.005 microcurie or more of removable alpha contamination, the licensee shall immediately withdraw the sealed source from use and shall cause it to be decontaminated and repaired by a person appropriately licensed to make such repairs or to be disposed of in accordance with the Commission regulations. Within five (5) days after determining that any source has leaked, the licensee shall file a report with the Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, describing the source, the test results, the extent of contamination, the apparent or suspected cause of source failure, and the corrective action taken. A copy of the report shall be sent to the Director of the nearest NRC Inspection and Enforcement Office listed in Appendix D of Title 10, Code of Federal Regulations, Part 20.

D. The periodic leak test required by this condition does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another person unless they have been leak tested within six (6) months prior to the date of use or transfer.

Annex B

PLUTONIUM ALPHA SOURCES

A. The sources when not in use shall be stored in a closed container adequately designed and constructed to contain plutonium which might otherwise be released during storage.

B. At least once every three months the licensee shall test the source for loss of plutonium in one of the following ways, using radiation detection instrumentation capable of detecting 0.005 microcurie of alpha contamination:

(1) By measurement of the source of potential alpha contamination through surveys of the storage container and areas in which the source is used; or

(2) By wiping thoroughly the external surfaces of the source mount, other than the radioactive surface of the source, with a piece of filter paper of high wet strength and low porosity moistened with a solution which will not attack the mount and, after the paper is allowed to dry, measuring the radioactivity on the paper.

C. If any survey or measurement performed as required by B.(1) or B.(2) above discloses the loss of more than 0.005 microcurie of plutonium from the source, or if a source has been damaged or broken, the source shall be deemed to be losing plutonium. The licensee shall immediately withdraw it from use, and cause the source to be decontaminated and repaired, or disposed of in accordance with the Commission regulations. Within five (5) days after determining that any source has lost plutonium, the licensee shall file a report with the Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, describing the source, the test results, the extent of contamination and the corrective action taken. A copy of the report shall be sent to the Director of the nearest NRC Inspection and Enforcement Office, listed in Appendix D of Title 10, Code of Federal Regulations, Part 20.

D. Records of test results shall be kept in units of microcuries and maintained for inspection by the Commission.

E. Notwithstanding the periodic test required by Paragraph B, any plutonium alpha source containing not more than 0.1 microcurie of plutonium is exempted from the above requirements.

Annex C

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

U. S. Nuclear Regulatory Commission
Division of Fuel Cycle and Material Safety
Washington, D.C. 20555

July 1982

The instructions in this guide, in conjunction with Table 1, specify the radionuclides and radiation exposure rate limits which should be used in decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control is considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 1 prior to the application of the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:

   a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

   b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Administrator of the NRC Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

   a. Identify the premises.

   b. Show that reasonable effort has been made to eliminate residual contamination.

   c. Describe the scope of the survey and general procedures followed.

   d. State the findings of the survey in units specified in the instruction.

   Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[b c f]	MAXIMUM[b d f]	REMOVABLE[b e f]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm βγ/100 cm^2	15,000 dpm βγ/100 cm^2	1000 dpm βγ/100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.