70-1113

GE Nuclear Energy

Nuclear Fuel & Components Manufacturing
General Electric Company
P.O. Box 780, Wilmington, NC 28402
919 675-5000

February 6, 1991

Director
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. C. J. Haughney, Chief
Fuel Cycle Safety Branch
OWFN, Room 6D23, Mail 6H3

Dear Sir:

Subject: License Amendment Request (Revision 29)

References: 1) NRC License, SNM-1097, Docket 70-1113
2) Telephone Conversation, E. D. Flack, R. E. Wilson, T. P. Winslow, S. P. Murray, F. G. Welfare, and R. H. Foleck, 2/1/91

With reference to activities authorized by NRC License SNM-1097 at the General Electric Company Nuclear Fuel and Components Manufacturing (NF&CM) facility, GE requests approval for the enclosed page changes hereby submitted for Parts I and II of our current application.

This request is to permit the criticality safety analysis for storage of fuel assemblies packed in RA containers to include consideration of gadolinium as a neutron absorber. Our current schedule indicates the storage of fuel assemblies, which must include this consideration, will be required by April 26, 1991.

Attachment 1 contains an explanation of the requested changes by page. Attachment 2 contains the revised page changes to the license conditions (Part I) and the demonstration (Part II) sections of our current application.

Please contact me at 919*675-5461 if you would like to discuss this matter further, or if you anticipate approval cannot be met by April 26, 1991.

Sincerely,

GE NUCLEAR ENERGY

T. Preston Winslow, Manager
Licensing & Nuclear Materials Management

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Attachments
cc: TPW-91-018

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ATTACHMENT 1

DESCRIPTION OF REQUESTED REVISIONS

Page(s)	Section	Description
I-1.14	1.8.4.3	Revised to limit the array height of RA inners containing rods/assemblies at reactor sites to comply with the criticality safety requirements.
I-4.12 4.13 4.14 4.15	4.2.4.4	Revised to describe the use of neutron absorber (gadolinium) mixed with fissile material to be used for geometry control. Also, to identify credit that may be taken for neutron absorbers that are normal constituents of filter media.
I-4.16 thru 4.26		Only the page numbers have changed due to the revisions on pages I-4.12 through I-4.15.
II-16.48 16.49 16.50 16.50.1 16.50.2 16.50.3 16.50.4 16.50.5 16.51	16.12	Revised to include the description for design process and fabrication controls of fuel assemblies. Pages II-16.48 through II-16.51 of the existing demonstration dated 9/30/83 should be removed and replaced with pages II-16.48 through II-16.51 dated 2/6/91.

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greater than those in the design bases for the containers during transportation.

Arrays can be constructed without limit to the number of containers so stored, except that each array shall be stacked to the smaller of 4 containers high or the height demonstrated to comply with criticality safety requirements. Each container must be separated by nominal 2 inch wooden studs, and with the width and length for each array and separation between arrays determined only by container handling requirements. *

Provisions for compliance with applicable 10 CFR 73 requirements are described in the NRC-approved GE-Wilmington Physical Security Plan dated June 6, 1986, as currently revised in accordance with regulatory provisions.

Storage at nuclear reactor sites is subject to the financial protection and indemnity provision of 10 CFR 140 and is limited to possession for purposes of delivery to a carrier for transport. The requirements of 10 CFR 70.24 are waived insofar as this section applies to the materials contained in any of the inner metal containers of the RA-series shipping package. (Reference Section 1.8.7).

1.8.4.4 Authorization to transfer, possess, use and store unirradiated reactor fuel of GE manufacture at nuclear reactor sites, for purposes of inspection, fuel bundle disassembly and assembly, including fuel rod and if, when accident conditions are credible, it also satisfies the accident condition specifications of Section 4.2.2.3.

4.2.4.3 Whenever criticality control is directly dependent on the integrity of a structure used to retain the geometric form of a fissile material accumulation or the spacing within a storage array, the structure shall be designed with an adequate strength factor to assure against failure under foreseeable loads or accident conditions. Materials of construction shall be fire resistant. The degree to which any corrosive environment might affect nuclear safety shall be considered and corrosion-resistant materials or coatings applied as necessary.

4.2.4.4 Neutron absorbers may be used as part of a defined geometry control when added to systems in one of the following ways. *

4.2.4.4.1 Fixed neutron absorbers may be considered part of a geometry control subject to the following: *

- The fixed neutron absorber is one of the following: *
  - Elemental cadmium
  - Elemental boron alloyed with steel
  - Solid, stable boron compounds such as boron carbide fixed in a matrix such as aluminum or polyester resin

- The fixed neutron absorber and any hydrogeneous material used to thermalize neutrons are sealed in a stainless steel or other suitable container which may be an integral part of the process equipment. *

- The fixed neutron absorber system (i.e., absorber, moderator, container) is installed as a permanent part of the process equipment such that it cannot be readily removed. *

- Prior to using a fixed neutron absorber system for criticality control, an inspection shall be performed using written procedures developed and approved by management to verify the presence of the neutron absorbers and to verify that the system installation is in accordance with design and nuclear criticality safety requirements. Inspection records shall be documented and maintained for the life of the system.

- The void volume of the fixed neutron absorber system is negligibly small to prevent internal rearrangement. *

- The effectiveness of the fixed neutron absorber system is demonstrated utilizing validated calculational methods. *

- The fixed neutron absorber system must not lose its effectiveness due to credible fire hazards. *
*

* The integrity of the fixed neutron absorber system must be verified on a periodic schedule compatible with the rates of corrosion and deterioration credible to the process system.

* If the fixed neutron absorber is removed, the system shall be locked out in accordance with safety lock and tag procedures (e.g., circuit switch locked, circuit breaker removed, feed lines removed) or operated under controls authorized in writing by the criticality safety function which ensure, as a minimum, that the requirements of Section 4.1.1 are met. *

* The fixed neutron absorber system must be designed so as not to lose its effectiveness due to credible industrial accidents and natural events. *

4.2.4.4.2 A neutron absorber admixed with fissile material may be used as part of a geometry control subject to the following: *

* The absorber is either elemental gadolinium or a solid stable compound of gadolinium such as Gd_2O_3. *

* The effect of the neutron absorber is considered as part of a criticality safety control only for those operations following the production of assembled fuel bundles. *

- Quality assurance requirements are in place which ensure that pellets, rods, and bundles contain minimum specified quantities of the absorber in the specified locations. *

- The on-site use of the neutron absorbing effect of admixed neutron absorbers is in accord with the provisions of Section 4.2.4.1. *

- The effectiveness of the admixed neutron absorber is demonstrated using validated calculational methods. *

4.2.4.4.3 Credit may be taken for neutron absorbers that are normal constituents of filter media (e.g., boron). Such credit may be taken only where the failure or loss of the media prevents accumulation of significant quantities of fissile material and where the absorber content of the media is certified. *

4.2.4.5 Whenever criticality control is directly dependent on the integrity of physical barriers or neutron absorbers, the structure shall be designed to protect against loss of integrity through foreseeable accident conditions such as fire, impact, melting, corrosion or leakage of materials. *

4.2.4.6 Where control of the spacing and/or height of movable units is used to provide criticality safety, the geometry of the system is administratively controlled by one or more of the following techniques:

- The geometry of the system is defined by engineered devices that determine the location of the movable units.

- The geometry of the system is defined by procedure through the use of visually identified storage areas.

- The geometry of the system is defined by engineered devices that limit the height of fissile material.

- The geometry of the system is defined by procedure through the use of maximum height restrictions.

Where this type of control is used, movable units must be safe geometry and must contain less than a minimum critical mass based on the form of the fissile material (i.e., powder, pellets or rods).

4.2.5 Nuclear Criticality Safety Considerations for Administrative Control of Mass

4.2.5.1 Where control of mass is used to provide criticality safety, the mass of uranium (or U^{235} or U^{238}) is administratively controlled based on measurement by one or more of the following techniques:

- The mass of uranium (or U^{235} or U^{238}) is determined as the product of the volume and the uranium (or U^{235} or U^{238}) concentration as measured by qualified counting methods.

- The mass of uranium (or U^{235} or U^{238}) is determined by qualified counting methods.

- The total mass or change in mass of a system is measured assuming the most reactive credible composition.

4.2.5.2 The use of mass control in criticality safety except as specifically licensed in Section 4.2.1, shall comply with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. That is, a control(s) must exist on another parameter(s) such that failure of the mass control to the most reactive credible value will not violate the multiplication limit for accident conditions specified in Section 4.2.2.3.

4.2.6 Nuclear Criticality Safety Considerations for Administrative Control of Moderation

4.2.6.1 Criticality safety of vessels, structures or processes may be based on control of moderation provided that the following conditions are satisfied:

- Sources of moderation internal and external to the process shall be identified and controls established for each source which are consistent with the requirements of Section 4.1.1.

- Support equipment associated with the control or processing of moderating materials shall be designed so that they are either geometrically safe or designed to prevent backflow of fissile materials and/or flooding of the fissile materials.

4.2.6.2 Where control of moderation is used to provide criticality safety, the degree of moderation is limited by one or more of the following techniques:

- Moderation is removed by an engineered system and the resulting material is inspected and sampled to ensure proper functioning of the equipment.

- Moderation is added procedurally and controlled by limiting the mass and/or volume of the moderator and fissile material.

- Moderation of the mixture is determined by analysis or engineered methods prior to using moderation as a criticality control.

4.2.6.3 The control of moderation for purposes of criticality safety must comply with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. That is, a control(s) must exist on another parameter(s) such that failure of the moderation control to the most reactive credible value will not violate the multiplication limit for accident conditions specified in Section 4.2.2.3.

4.2.7 Criticality Safety Requirements for Use of Concentration Control

4.2.7.1 Where control of uranium concentration is used to provide criticality safety, the concentration is controlled by one or more of the following techniques:

- The solubility is controlled to prevent precipitation and the material is either agitated or recirculated at a rate sufficient to prevent settling into an unsafe concentration.

- The uranium concentration is limited by on-line measurement of concentration (or density if the worst credible composition is assumed). If the limit is reached, automatic controls must prevent continued increase.

- The uranium concentration in a precipitate is measured and limited assuming the worst credible composition.

4.2.7.2 A full density mixture is used in determinations of uranium concentration (i.e., the effect of voids or inert materials mixed with the accumulation is not included).

4.2.7.3 The control of concentration for purposes of criticality safety, except as specifically licensed in Section 4.2.1, shall comply with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. That is, a control(s) must exist on another parameter(s) such that failure of the concentration control to the most reactive credible value will not violate the multiplication limit for accident conditions specified in Section 4.2.2.3.

4.2.8 Criticality Safety Requirements for Use of Density Control

4.2.8.1 Where control of uranium density is used to provide criticality safety, the uranium density is administratively controlled by one or more of the following techniques:

- Fuel displacing media is maintained within the geometry by use of mechanical attachments or by use of a size/shape that cannot exit from the geometry during use. Where this method of criticality control is used, it is necessary to assume the worst credible distribution of media within the geometry.

- The uranium density is controlled by use of homogenized mixtures with measured uranium density.

Where materials having different uranium densities are mixed, the maximum density measured is assumed throughout the system.

4.2.8.2 The control of density for purposes of criticality safety must comply with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. That is, a control(s) must exist on another parameter(s) such that failure of the density control to the most reactive credible value will not violate the multiplication limit for accident conditions specified in Section 4.2.2.3.

4.2.9 Criticality Safety Requirements for Use of Enrichment Control

4.2.9.1 In the controls of Sections 4.2.4 through 4.2.8, a maximum area enrichment is implicit in the analysis. However, for some situations in a limited portion of an area, it is desirable to control enrichment to a lower value. Where control of enrichment to a value lower than the maximum area value is used to provide criticality safety, the U^{235} enrichment is administratively controlled based on measurement by standard assay techniques prior to enrichment being used as a criticality control. Where materials having different enrichment values are mixed, the maximum enrichment measured is assumed throughout the system.

4.2.9.2 The control of enrichment for purposes of criticality safety to a value lower than the maximum area value must comply with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. That is, a control(s) must exist on another parameter(s) such that failure of the enrichment control to the maximum area value will not violate the multiplication limit for accident conditions specified in Section 4.2.2.3.

4.2.10 <u>Criticality Treatment of Interaction between Subcritical Units</u>

4.2.10.1 Equipment and facilities may be considered to be nuclearly isolated if they are separated by either of the following:

- A one foot slab of water or by the distance which is equivalent in isolation ability to a one foot slab of water.

- The larger of 12 feet or the greatest distance across an orthographic projection of the largest of the fissile accumulations on a plane perpendicular to the line joining their centers.

4.2.10.2 The criticality effects of the exchange of neutrons between individual subcritical units which are not isolated may be treated by either of the following techniques:

- Techniques which produce a calculated multiplication factor of the entire system (e.g., Monte Carlo) may be used. When this is done, the analysis must comply with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4.

- Techniques which do not produce a calculated multiplication factor for the entire system but instead compare the system to accepted criteria (e.g., Solid Angle technique) may be used. When this is done, the analysis must comply with the requirements of Sections 4.1.1 and 4.2.2.4.

4.2.11 Requirements for Engineered Controls

Engineered controls detect an undesired situation and implement corrective action without requiring human intervention. Engineered controls must be:

- Sufficiently dependable and so used that the probability of failure is minimized.

- Capable of performing the criticality safety purpose for which they are specified.

- Maintained and/or calibrated on a schedule suitable for the specific device and the specific application.

- Verified as being properly installed prior to first use with fissile material.

- Modified only with documented, prior approval of the criticality safety function.

- Supported by procedures and/or devices which provide continued control if operation is to be allowed to continue after a control has failed.

- So designed that when sampling is part of a control, the sampling is performed at a frequency consistent with the rate of variation of the parameter and with the implementation of control action.

4.2.12 Requirements for Procedural Controls

,, procedural control requires human intervention in detecting an undesired condition and/or implementing corrective action. Procedural controls must be:

- Sufficiently dependable and so used that the probability of failure is minimized.

- Capable of performing the criticality safety purpose for which they are specified.

- Implemented by formal written procedures.

- Shown to be complied with by formal written records.

- Modified only with documented, prior approval of the criticality safety function.

- Supported by procedures and/or devices which provide contin ed control if operation is to be allowed to continue after a control has failed.

- So designed that when sampling is part of a control, the sampling is performed at a frequency consistent with the rate of variation of the parameter and with the implementation of control action.

4.2.13 Criteria for Fire Protection in Areas Containing Fissile Material

4.2.13.1 Fire protection shall be provided for equipment, processes, and facilities containing fissile material and shall be selected on the basis of minimum impact on area nuclear criticality safety.

4.2.13.2 The use of water for fire protection in moderation control areas shall be minimized and controlled.

4.2.13.3 Fire protection instructions covering the manufacturing facility are issued which communicate necessary or permissible methods or techniques to be used.

4.2.14 Incineration of Nuclear Waste

The incinerator is authorized to operate with an estimated uranium enrichment of not less than 2.75% in U^{235} for batch control. Waste boxes with assigned enrichments greater than 2.75% in U^{235} can be incinerated with proper demonstration of safety provided the boxes are generated from process areas where the maximum nominal enrichment handled is not more than

4.0%. No waste boxes generated from a process which operates with a maximum enrichment greater than 4.0% nominal will be incinerated unless physical measurements of the U^{235} content are made or the highest enrichment in the box is assigned to the batch.

Bundles are handled on a batch basis. The leak check and inspection stations are spaced at distances greater than that of the bundle storage rack described in the next section.

16.12 FUEL BUNDLE STORAGE AND PACKAGING *

16.12.1 Process Description of Fuel Bundle Storage *

Following the leak test, bundles are moved to a bundle storage rack or directly to a final inspection station, each fuel bundle is wrapped, but not sealed, with a plastic dust cover. *

The storage area consists of two rows of racks. Six of the original eight rows have been removed. However, one of the remaining two is currently blocked off and the safety analysis described below continues to support this arrangement. Each rack can hold 56 bundles on each side. The racks are rigidly constructed of steel girders on 48-inch centerlines. The center-to-center spacing of rows on the same rack is 16.75 inches. The center-to-center spacing of bundles in the same row is 15 inches. The center-to-center spacing of rows of bundles adjacent to each other and hanging on adjacent racks as presently used is 79.25 inches. *

16.12.2 Criticality Safety of Fuel Bundle Storage *

Criticality safety of the fuel bundle storage area is maintained by the fixed spacing provided by the storage rack. The fuel rods in the individual assemblies are in * fixed positions leaving well-defined air spaces within the assembly which could conceivably be occupied (at least partially) by water from overhead sprinklers. For this reason, optimum interspersed moderation is assumed to be possible within the interstices as well as between the assemblies. *

The criticality safety analysis for the fuel bundle storage area was performed with the GEMER Monte Carlo code. The model developed for the GEMER analysis consisted of one half of the array with the remainder approximated by reflecting boundary conditions. A maximum enrichment of 4.025 weight percent U^{235} was assumed for all bundles. The interspersed moderation was varied to evaluate the multiplication of the system at the optimum value. The following assumptions were made: a 19-inch concrete floor is located 20 inches below the bottom tie plate, water reflectors, 12.9 inches thick, on all sides of the array, and the top is reflected by 15.3 inches of concrete. The calculated K-effectives demonstrate the system to be acceptable under the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. *

When fuel assemblies are covered with plastic sleeves, the bottom of the sleeves will be left open to prevent buildup of water within the sleeves. *

The fuel assembly storage racks are rigidly constructed steel frameworks which securely hold the bundles in their designated positions. Loading of the rack with *

more than one assembly per storage position is *
physically impossible. *

Interaction of the storage area with special nuclear *
material accumulations in adjacent areas was analyzed by *
the solid angle method and determined to be within *
acceptable limits. *

16.12.3 <u>Process Description of Packaging of Fuel Bundles and Loose Rods for Transport</u> *

Released fuel bundles are packed with plastic spacers *
between rods in each direction to prevent rod movement *
during transportation. Bundles are removed by an *
overhead crane to the inner metal containers of the GE *
RA-series shipping package. This container is designed *
to hold no more than two bundles and to be loaded in the *
vertical position. The loaded container is then turned *
into the horizontal position, sealed, and placed in the *
outer GE RA-series shipping container. The cover of the *
outer container is bolted in position. *

According to a predetermined shipping schedule, the *
containers are transferred to the transport vehicle for *
shipment to an authorized receiver. *

Individual fuel rods are packaged for shipment in the RA *
series shipping container. Each rod is placed in a *
plastic sleeve with up to 70 rods being bound together *
as a group. Fuel rods with an average U^{235} enrichment *
greater than 3.2% are placed in a schedule 40 stainless *
steel pipe. Up to two groups of rods are loaded in each *
metal inner RA container for shipment. *

16.12.4 Criticality Safety of Packaging *

RA Containers are lined with shock absorbent material. Packaging operations are conducted in compliance with 10 CFR Part 71. The RA-series shipping package capacity is limited to two bundles. The container has been analyzed using the GEMER code with an explicit generalized geometry description of the fuel bundles and the materials of the container. The effect of neutron absorption in gadolinium was not considered in this analysis. This analysis demonstrates that subject to restrictions on the height of arrays and for enrichments up to 4.025% U^{235} the use of the container at GE-Wilmington complies with the requirements of Sections 4.1.1, 4.2.2.3 and 4.2.2.4. *

16.12.5 Neutron Absorbers in Fuel Bundle Storage and Packaging *

Neutron absorbers may be considered in demonstrating compliance with sections of 4.1.1, 4.2.2.3, and 4.2.2.4 of Part I subject to the provisions of 4.2.4.4.2. *

The regulation for assuring quality is established in 10 CFR 50, Appendix B, and our nuclear safety function has determined that its provisions are adequate to assure the presence of proper enrichment and gadolinia content. GE-Nuclear Energy's quality assurance program designed to comply with this regulation (10 CFR 50, Appendix B) is identified in the NRC approved "GE Nuclear Energy Quality Assurance Program Description, NEDO-11209-04A, Revision 8, March 31, 1989. *

The current quality control features that provide assurance for the presence of proper enrichment and gadolinium content are provided below. *

16.12.5.1 Controls at the Fuels Bundle Design Stage *

This section describes the controls which are employed at the assembly design stage to ensure that approved designs contain the minimum amount of gadolinium at the prescribed locations, per the RA container reactivity control requirements. The quality assurance controls on the manufacturing process are discussed in Section 16.12.6.2. *

The nuclear design of the fuel assembly determines the average enrichment and gadolinia loading for each lattice in the assembly. As will be described below, the RA container requirements governing the nuclear design of the fuel are met by this design process. *

The nuclear design is controlled by a Design Basis Document. This document contains all the key requirements for the design, including the RA container requirements. Because the nuclear designer must conform to all the requirements of this document, the RA container requirements become an integral part of the design process. Therefore, the restraints on the average lattice enrichment and minimum gadolinia content and location of gadolinia rods in the assemblies are integrated into the design from the outset. *

When the design is completed, it must be independently verified before it can be applied. An independent verification is performed to ensure that the design meets all the requirements specified in the Design Basis Document. The standard checklist for the verification includes verification that the fuel assembly design meets RA container Certificate of Compliance criteria authorized by the NRC.

After the design has been verified, and found to meet all requirements, it is then released to the Engineering Data Bank (EDB). The EDB is a computer based databank governed by procedures which ensure the integrity of the data which it contains. Selected data from the file created by nuclear engineering (including rod-by-rod enrichment and gadolinia loadings), plus other pertinent mechanical design data, is then combined to create engineering drawings of the assembly for use by manufacturing. In addition, this same data is used to create a separate file on the EDB which is electronically transmitted to manufacturing for use in the automated assembly of the fuel. Both the drawings and data files are independently verified before transmittal to manufacturing.

The design process outlined above ensures that the RA container requirements are met by all assemblies designed by nuclear engineering and released to manufacturing.

16.12.5.2 Controls at the Fuel Bundle Fabrication and Assembly Stage *

The Automated Fuel Release Design System (AFRDS) is a computer system tied into the Engineering Data Bank (EDB). AFRDS electronically transmits fuel assembly design data to manufacturing. The Automated Design and Manufacturing System (ADAMS) electronically receives data from AFRDS. This system converts data into a usable format for Manufacturing and Quality overchecks. This data is verified by our Quality organization for agreement with the issued Design Engineering drawings and parts lists. The verifications include specific fuel rod and assembly requirements, fuel assembly serial numbers and fuel rod type assignments. Upon satisfactory completion of the ADAMS verification, the project data is downloaded to other computer user systems. *

Manufacturing generates Project Information Sheets (PIS) from Design Engineering drawings to detail the individual fuel rod requirements for enrichment, gadolinia content and fuel zone lengths. Fuel rods are fabricated according to the requirements of the PIS sheets. The Material Inventory Control System (MICS) receives downloaded ADAMS data and verifies the fuel rods are being loaded to specification requirements. UO_2 fuel enrichment, gadolinia content and other fuel specification requirements are verified on a sampling basis via qualified analytical techniques prior to rod loading. *

All completed UO_2 and gadolinia fuel rods are inspected for U_{235} content, enrichment, gadolinia weight % and zone positions via fuel rod scanners prior to being released for fuel bundle assembly. ADAMS and MICS data are used to generate the zone by zone requirements for each fuel rod type to be scanned. This data is used by the scanners to overcheck fuel rod conformance to design specifications. *

The fuel rod assignment to a location within an assembly and the subsequent placement within that assembly location is assured by two independent automated optical serial number readers attached to separate computer systems. One of these systems is at bundle rod accumulation and the other is at the Automated Bundle Assembly Machine (ABAM). *

ABAM receives bundle rod type and location information from MICS. ABAM reads fuel rod serial numbers and accepts only rods which have been approved for assembly by MICS. ABAM then assigns the released rod to the authorized design location in the assembly and the rod is automatically inserted into that location. *

The Fuel Certification System (FCERT) database is received from ADAMS and checked by Quality for transmitted errors. The FCERT assures that all fuel rod and bundle data exhibit no errors. The specific bundle requirements in FCERT, which include inspections during assembly and packaging, must be completed prior to certification of the fuel bundle. *

16.13 This section has been left blank intentionally. *

16.14 SCRAP RECOVERY

Internally-generated uranium as compounds in various physical forms which do not meet quality standards, or which have been mixed with foreign material, is reprocessed through scrap recovery equipment. This equipment is located in the UF_6-to-UO_2 conversion area. Recovered material is later blended at appropriate points in the process with primary production flow.

16.14.1 Dissolution & Filtration

Material to be reprocessed in scrap recovery equipment is accumulated in batches in 5-gallon pails at the various locations where scrap is generated. A storage area for accumulating pails of scrap is provided.

One product pail containing a dissolver batch is placed in a ventilated cabinet where the lid is removed.

The pail is then transferred into position over the charging chute of a dissolver. Nine 10" diameter vertical stainless steel dissolvers are arranged in three sets of three. The dissolver has been previously charged with nitric acid to which the scrap is slowly added and the contents heated with steam to 180°F and