UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

RC File

B-1-A

NUCLEAR FUEL SERVICES, INC.
AND
NEW YORK STATE ATOMIC AND SPACE DEVELOPMENT AUTHORITY

DOCKET NO. 50-201

PROVISIONAL OPERATING LICENSE

License No. CSF-1

The Atomic Energy Commission having found that:

a. The application for license complies with the requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations set forth in Title 10, CFR;

b. Construction of the facility has been completed in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission;

c. As an intermediate procedure prior to issuance of an operating license pursuant to Title 10, CFR, § 50.56, a provisional operating license should be issued because there are involved features, characteristics or components of the proposed facility as to which it appears desirable to obtain actual operating experience before issuance of an operating license for the full term requested in the application;

d. There is reasonable assurance (i) that the activities authorized by the provisional operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in Title 10, CFR;

e. The applicants are technically and financially qualified to engage in the activities authorized by the provisional operating license in accordance with the regulations in Title 10, CFR;

f. The applicable provisions of Part 140, Title 10, CFR, have been satisfied; and

g. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

Provisional Operating License No. CSF-1 is hereby issued as follows:

1. This license applies to the irradiated nuclear fuel processing plant (the "facility") located at the Western New York Nuclear Service Center, Cattaraugus and Erie Counties, New York, and described in

Part B of the revised license application, as amended, filed by Nuclear Fuel Services, Inc. ("NFS") and incorporated by reference in the revised license application, as amended, of the New York State Atomic and Space Development Authority ("ASDA"). Volumes 1 and 2 of said Part B of the revised license application, as amended, entitled "Final Safety Analysis Report" are considered to be the "Hazards Summary Report".

2. Subject to the conditions and requirements incorporated herein the Commission hereby licenses NFS -- as lessee of the site; as owner of those portions of the facility in which actual chemical processing will take place; and as the party responsible for the operation of the facility, including storage of irradiated fuel elements, storage of radioactive wastes and burial of radioactive waste --

   A. Pursuant to Section 104.b of the Atomic Energy Act of 1954, as amended (the "Act"), and Title 10, CFR, Part 50, "Licensing of Production and Utilization Facilities", to possess, use and operate the facility as a production facility;

   B. Pursuant to the Act and Title 10, CFR, Part 70, "Special Nuclear Material", to receive, acquire, possess and use that amount of special nuclear material in or from irradiated solid nuclear fuel elements, and that amount of special nuclear material contained in auxiliary sources such as calibration and laboratory standards, the receipt, storage or processing of which is authorized by Section 2 of the technical specifications appended to this license;

   C. Pursuant to the Act and Title 10, CFR, Part 30, "Licensing of Byproduct Material", to receive, separate, possess and use that amount of byproduct material in or from irradiated solid nuclear fuel elements, and to receive, possess and use that amount of byproduct material contained in auxiliary sources such as calibration and laboratory standards, the receipt, storage or processing of which is authorized by Section 2 of the technical specifications appended to this license;

   D. Pursuant to the Act and Title 10, CFR, Part 40, "Licensing of Source Material", to receive, possess and use that amount of source material in or from irradiated solid nuclear fuel elements, and that amount of source material contained in fuel element prototypes and other auxiliary forms, the receipt, storage or processing of which is authorized by Section 2 of the technical specifications appended to this license; and

E. Pursuant to the Act and Title 10, CFR, Parts 20, 30, 40, and 70 to dispose of solid radioactive waste generated in the operation of the facility by burial in the soil in accordance with the technical specifications.

3. Subject to the conditions and requirements incorporated herein the Commission hereby licenses ASDA -- as owner and lessor of the site, of those portions of the facility in which the preprocessing storage of irradiated fuel elements and the storage and burial of radioactive wastes will take place, and of other site improvements -- to possess title to those portions of the facility not owned by NFS and to permit NFS to perform those acts which NFS is authorized to do by paragraph 2 of this license.

4. A. Notwithstanding any expiration, modification, cancellation or termination of the contractual arrangements between NFS and ASDA, NFS shall, so long as this license shall be in force, with respect to NFS, be responsible for assuring that the provisions of this license and Commission regulations for protection of health and safety from radiation hazards are observed with respect to the facility and materials covered by this license. In the event of any expiration, modification, cancellation or termination of the contractual arrangement between NFS and ASDA or any other change in the relationship between them, including any proposed transfer from NFS to ASDA of responsibility for the operation and care of those portions of the facility in which the storage and burial of radioactive wastes will take place, NFS or ASDA may apply to the Commission for an appropriate amendment of this license reflecting the future responsibilities of NFS and ASDA with respect to satisfying Commission regulatory requirements. Until such amendment is issued, ASDA shall in no way prevent NFS from observing the requirements set forth in this condition.

B. To the extent that the operation of the facility under this license results in the production of radioactive wastes to be stored in portions of the facility or in improvements hereafter constructed at the site, or otherwise to be managed at the site, beyond the term of this license or any superseding license, NFS or ASDA may apply to the Commission for an appropriate amendment of this license or any superseding license with respect to such continued storage or management in accordance with Commission regulations.

5. Except as specifically otherwise provided by the Commission, this license shall be deemed to contain and be subject to the conditions specified in Section 50.54 of Part 50, Section 70.32 of Part 70, Section 40.41 of Part 40 and Section 30.34 of Part 30 of the Commission's regulations; is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect, including Part 20; and is subject to the additional conditions specified below:

A. Technical Specifications

The technical specifications for operation of the facility contained in Appendix A attached hereto are hereby incorporated in this license. Except as hereinafter provided, the facility shall be operated in accordance with the technical specifications. Changes may be made in the technical specifications only when authorized by the Commission in accordance with the provisions of Section 50.59 of the Commission's regulations (Title 10, CFR, Part 50, "Licensing of Production and Utilization Facilities").

B. Records

In addition to those otherwise required under this license and applicable regulations, NFS shall keep the following records:

(1) Records showing the radioactivity released or discharged to unrestricted areas as measured at or prior to the point of such release or discharge.

(2) Records of radioactivity measurements at on-site and off-site monitoring stations described in the technical specifications.

(3) Records of the radioactive material received, transferred, stored as high-level liquid waste or disposed of as solid waste.

(4) Records of facility tests, measurements and calculations performed pursuant to the requirements of the technical specifications.

(5) Records of major repairs or substitution or replacement of major equipment listed in Equipment List, Appendix 5.2 of the Final Safety Analysis Report.

(6) Letters of authorization, run sheets, operating logs and check-off lists of the mechanical processing and chemical processing operations performed in the plant.

NFS is hereby authorized to dispose of the records described in 5.B.(4), (5) and (6) after a retention period of two years from the date of occurrence of the activity recorded. Records described in 5.B.(1), (2) and (3) shall be retained until disposal is authorized by the Commission.

C. Reports

In addition to reports otherwise required by this license and applicable regulations:

(1) Any operation outside the limits established by the technical specifications, and any operation of the facility in the "ready condition" as defined in the technical specifications, shall be promptly reported by telephone or telegraph to the Director of the appropriate Atomic Energy Commission Regional Compliance Office listed in Appendix D of 10 CFR 20. NFS shall submit within 10 days a complete report of the causes and corrective actions taken. This report shall be submitted to the Director, Division of Materials Licensing, with a copy to the Regional Compliance Office.

(2) NFS shall report to the Commission in writing within 30 days of its occurrence any change in the plant organization indicated in the amendment to the application by NFS dated October 19, 1965.

(3) NFS shall report to the Commission in writing within 30 days of the time it is observed any change in the validity of the assumptions used in the accident analyses, as described in Section VII of the Final Safety Analysis Report.

(4) NFS shall submit to the Commission, at least quarterly, during the period of this provisional operating license, a written report covering the following matters:

a. A summary of the processing operations performed during the period, including their duration.

b. The amount of radioactive material received, transferred, stored as high-level liquid waste, disposed of as solid waste, and released as liquid, gaseous and solid effluent.

c. The levels of radioactivity measured at the stack, the site-perimeter monitoring stations and in the milk from the cows on the NFS-operated farm.

d. A brief explanation of the cause of each unplanned process shutdown.

e. A description of major repairs performed in the facility with reasons therefor.

f. A description of changes, tests, and experiments performed pursuant to Paragraph 50.59(a) of the Commission's rules and regulations.

g. A description of malfunctions of any equipment listed in Appendices 5.2, 9.51, 9.53, and 9.56 of the Final Safety Analysis Report which is important to safety.

h. The results of periodic testing performed in accordance with Section 6 of the technical specifications.

Such reports shall be submitted within 30 days after the end of each reporting period.

6. This license is effective as of the date of issuance and shall expire eighteen months from the date of issuance (unless extended for good cause shown), or upon the earlier issuance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
L. A. McBride

Director
Division of Materials Licensing

Attachment:
Appendix A

Date of Issuance: APR 19 1966

APPENDIX A

NUCLEAR FUEL SERVICES, INC.
AND
NEW YORK STATE ATOMIC AND SPACE DEVELOPMENT AUTHORITY

IRRADIATED NUCLEAR FUEL PROCESSING PLANT

TECHNICAL SPECIFICATIONS
LICENSE CSF-1

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1.0 PLANT DESCRIPTION SUMMARY

Details of the plant, operation, protection of the public, etc., may be found in the Final Safety Analysis Report submitted as a part of the license application for the NFS Spent Fuel Processing Plant.

1.1 Location of Site

The NFS Spent Fuel Processing Plant is located at the Western New York Nuclear Service Center, a 3331-acre site in the town of Ashford in the north central section of Cattaraugus County (14 acres in Concord Town, Erie Co.). The boundaries of the site superimposed on a topographic map of the area are shown in Fig. 2.7a of the Final Safety Analysis Report. The plant is located near the center of the site on a mesa-like peninsula, bordered on the east by Frank's Creek and on the west by Quarry Creek. These two defiles are deep enough that, considering the water table contours, it can be definitely stated that any activity getting into the ground water in the plant site area will show up eventually in one of these two streams and nowhere else, except, of course, for that which is sorbed upon the soils and held therein. This situation permits comprehensive monitoring of liquid effluents which may contain radioactivity.

1.2 Location of the Plant

The plant is located near the center of the Service Center (Fig. 3.1a, ibid) and is separately fenced with an 8-foot high exclusion fence. The plant is about 1200 meters from the nearest site boundary. The building contains about 80,000 square feet of gross floor area. The process areas have been grouped together as much as possible to minimize piping runs and to provide a reasonable flow of material from the introduction of the fuel into the plant to the shipment of purified products.

1.3 Flow of Material

The normal flow of material through the plant follows: Spent fuel elements are received in casks by rail or truck into the Fuel Receiving and Storage Area (FRS). The cask temperature and pressure are checked and, if necessary, after radiation monitoring, external decontamination of the cask is carried out. The cask is next placed into a 44-foot deep cask unloading pool, the cover removed and the elements placed into storage canisters. These canisters are then transferred to the fuel storage pool for storage prior to further processing.

The next sequential operation is performed in the Process Mechanical Cell (PMC). The storage canisters are moved one at a time to the underwater transfer conveyor and the fuel assemblies (only) introduced into the PMC. Here, the fuel elements are mechanically disassembled and chopped, if required, to prepare them for dissolution. The prepared elements are

placed in baskets, stored in the General Purpose Cell (GPC), and are subsequently charged to the dissolver located in the Chemical Processing Cell (CPC).

Dissolution is carried out using nitric acid and the dissolver solution is transferred by steam jet to the accountability and feed adjustment tank also located in the CPC. After analysis and adjustment, the feed is jetted to the partition cycle feed tank located in the first extraction cell (XC1).

Countercurrent solvent extraction column techniques are employed to separate fission products from the uranium and plutonium and, subsequently, to separate the uranium and plutonium. After initial decontamination the uranium-bearing solution undergoes two further solvent extraction purification cycles; the plutonium-bearing solution, one. After leaving the solvent extraction columns, the uranium-bearing solution undergoes an additional purification step by means of silica gel bed sorption, the plutonium-bearing solution by ion-exchange techniques. Product solutions are concentrated by evaporation, packaged in geometrically safe containers and placed in birdcages, when necessary, for storage and shipment.

1.4 Ancillary Systems

Additional systems provide for rework of off-specification process materials, off-gas treatment, acid recovery, solvent recovery, plant ventilation, temporary holdup of liquid effluents, underground tank storage of radioactive waste and solid radioactive waste burial.

2.0 POSSESSION LIMITS AND FORM OF MATERIALS

2.1 Possession Limits

The possession limits of U-235, Pu and U-233 for the Nuclear Fuel Spent Fuel Processing Plant shall be as follows:

U-235	21,000 kg
Pu	4,000 kg
U-233	3,200 kg

These values are based on the following assumptions:

(a) that the 924 canister spaces in the fuel pool are filled with canisters each of which contain 10 kg of U-235 in addition to the Pu and U-233 which could be produced by a 40% burnup of this quantity of U-235 at a conversion ratio of 0.8;

(b) that all product storage tanks are filled with solution at a maximum concentration of U-235 as specified in Section 4.8.1.1 as low enriched uranium or 360 g/l of plutonium or U-235 as high enriched uranium; and

(c) that all birdcage storage spaces are occupied with birdcages containing solutions at a maximum concentration of 360 g/l of U-235, Pu or U-233.

2.2 Form of Materials

The special nuclear material (SNM) in the fuel storage pool shall be in those forms listed in the following table-Enumeration of Material Categories; in other parts of the plant the material may be in those forms required for the flow of material as described in Section 1.3 of these Technical Specifications or in related research and/or development work, provided, however, that no conversion of fuels of Categories III, V, and VII from the form in which they are received may be made in the plant until such time as installation of the tankage which may be necessary for storage of the processing waste from these fuels has been completed and approved by the United States Atomic Energy Commission.

2.3 Laboratory, Calibration and Test Materials

Certain materials, including SNM not covered in Sections 2.1 or 2.2, of the following types and quantities may be received, possessed, stored and used as laboratory or calibration standards or for tests and measurements.

Material	Form	Possession Limit
U-233	Nitrate solution	6.0 g U-233
Plutonium-242	Nitrate solution	6.0 g Pu-242
Plutonium	Sulfate hydrate NBS Isotopic Standard 948	5.0 g Pu
Plutonium	Metal NBS Standard 949	10.0 g Pu
Uranium	Uranium Oxide NBS Isotopic Standards 010 to 930	50 g U-235
Uranium	Uranium Oxide NBS Standard 950 a	3.0 g U-235
Uranium-Plutonium Mixture	UO_2 + PuO_2	0.5 g Pu 0.9 g U-235
Americium-241	Deposited NBS Source 4904	1.1×10^{-9} curie
Polonium-210	Deposited NBS Sources 4900, 4901, 4902	5×10^{-8} curie
Manganese-54	NBS Point Source 4977 B	1.4×10^{-6} curie
Zinc-65	NBS Point Source 4992 D	1.4×10^{-6} curie
Zinc-65	NBS Solution Standard 4903	1×10^{-5} curie
Strontium-90	Sealed Source	5×10^{-6} curie
Pu-239	Sealed Source	1×10^{-6} curie
U-Th mixtures from 1 to 3½% enriched	UO_2 and ThO_2	50 g
Co-60	Sealed Source(s)	500.05 curie
Cesium-137	Chloride Solution	5×10^{-6} curie
Iodine-131	Solution	5×10^{-6} curie
Iodine-131	Elemental	5.1×10^{-2} curie
H^3	Tritium Solutions	1×10^{-3} curie
Carbon-14	Solutions	1×10^{-5} curie

2.4 Source Material

Source material of the following types and quantities may be received, possessed, stored and used in conjunction with the operation of the facility.

Material	Form	Possession Limit
Natural Uranium	Metal-NPR type fuel elements	50,000 lbs
Depleted Uranium	UO_2 + metal prototype fuel elements and U_3O_8 granules	100,000 lbs
Thorium	Thorium Nitrate	50,000 lbs

ENUMERATION OF MATERIAL CATEGORIES: PARTICULAR SPECIFICATIONS

CATEGORY	DETAILS
I.	UO_2 sheathed in stainless steel, zirconium, or Zircaloy. Enrichment of U-235 as initially charged to the nuclear reactor of all the uranium in the processing lot must not exceed 5%. The design of the assembly must be such that the assembly casing and end fittings are removable without undue difficulty with the equipment provided in the Process Mechanical Cell of the Plant. No individual fuel element may exceed 0.75" in diameter and fuel cladding thickness may not exceed 50 mils. Metallic hardware inside the section of the assembly containing source material or special nuclear material may not exceed 1/8" thickness. The fuel material must be essentially UO_2 prior to irradiation.
II.	The same specification as for Category I except that the initial U-235 enrichment as charged to the nuclear reactor of all or part of the contained uranium is more than 5% but all is less than 10%.
III.	UO_2ThO_2 sheathed in stainless steel, zirconium or Zircaloy. Uranium used in the fuel may be highly enriched with U-235 (93.5%) or lower enrichment. The design of the assembly must be such that the fuel assembly casing and end fittings are removable without undue difficulty with the equipment provided in the Process Mechanical Cell of the Plant. No individual fuel element may exceed 0.75" in diameter and fuel cladding thickness may not exceed 50 mils. Metallic hardware in the section of the assembly containing source material or special nuclear material may not exceed 1/8" in thickness. The fuel material must be essentially ThO_2 or ThO_2-UO_2 prior to irradiation. The uranium (elemental basis) content shall not exceed 8.5%.
IV.	U-Mo alloy contained in tubes of stainless steel, zirconium, or Zircaloy. Uranium used in the alloy must be no higher than 26.5% U-235 enrichment. The design of the assembly must be such that the assembly casing is removable without undue difficulty with the equipment provided in the Process Mechanical Cell of the Plant. No individual fuel element may exceed 0.75" in diameter and fuel cladding thickness may not exceed 50 mils. Metallic hardware in the section of the assembly containing source material or special nuclear material may not exceed 1/8" in thickness. The alloy must be essentially U-Mo prior to irradiation. The average molybdenum content for a processing lot may not exceed 10%.

CATEGORY	DETAILS
V.	U-Zr alloy or U-Zircaloy alloy clad in zirconium or Zircaloy and contained in plate-type fuel elements requiring no mechanical head-end treatment other than the removal of end fittings in order to be charged to the dissolver. Uranium used in the alloy may be highly enriched with U-235 (93.5%) or any lower enrichment. The alloy must be essentially U-Zr prior to irradiation. The uranium content of the alloy plus cladding must not exceed 10%.
VI.	UO_2 or uranium metal sheathed in aluminum and contained in plate, rod or hollow cylindrical type fuel elements. Initial U-235 enrichment of all uranium must not exceed 5%. The design of the assembly must be such that the assembly casing and end fittings are removable without undue difficulty with the equipment provided in the Process Mechanical Cell of the Plant. The fuel material, prior to irradiation, must be essentially UO_2 or U metal except for bonding materials such as Ni or Al-Si.
VII.	U-Al alloy clad in aluminum and contained in plate-type fuel elements requiring no mechanical head-end treatment other than the removal of end fittings. Uranium used in the alloy may be highly enriched with U-235 (93.5%) or any lower enrichment. No fuel element may be more than 4-1/2 inches on a side or 40 inches long in fuel length. The alloy must be essentially U-Al prior to irradiation. The uranium content of alloy plus cladding must not exceed 10%.
VIII.	U-Mo alloy clad in aluminum and contained in hollow cylindrical fuel elements. The design of stainless steel end fittings must be such that they are removable without undue difficulty with the equipment installed in the Process Mechanical Cell of the Plant and it must be possible to push the aluminum-clad fuel elements out of the stainless steel casing. Uranium used in the alloy must not exceed 4.5% in initial U-235 enrichment. The individual fuel element must not exceed 24 inches in length or 5.5 inches in outside diameter. The alloy must be essentially U-Mo prior to irradiation. The Mo content of the alloy may not exceed 4%. The assembly may contain up to 2.3% nickel based on the total weight of assembly.

CATEGORY	DETAILS
IX.	Uranium metal clad in zirconium or Zircaloy and contained in tubular fuel elements as specifically described in Drawing No. NPR-1, dated 3/27/63. Enrichment of U-235 as initially charged to the nuclear reactor of all the uranium in the processing lot must not exceed 1.5%. The fuel material must be essentially uranium metal prior to irradiation. It is noted, however, that silicon, iron and/or aluminum may be added to the uranium metal in amounts not exceeding 1000 ppm parts of uranium each.

3.0 DEFINITIONS

The following terms are defined only for the purpose of clearly indicating the intent of the various provisions given within these Technical Specifications.

Blanking Off	The insertion of a removable positive barrier in a line.
Boral	A sandwich-type plate with a core of boron carbide (B_4C) crystals suspended in cast aluminum, clad on both sides with aluminum.
Concentration Control	A technique used to ensure nuclear safety through control of the concentration of fissionable isotopes in process and product solutions.
Fixed Poison Tanks	A tank (or vessel) containing fixed neutron absorbers, i.e., borosilicate - glass Raschig rings or boron-stainless steel Raschig rings.
Flooding velocity	That velocity of one or both phases in a column at which the column is no longer capable of providing deentrainment of phases in the effluent streams.
Geometrically Favorable	A vessel which is safe by geometry for all credible concentrations of fissionable material.
Gross count	Total alpha, beta and gamma radioactivity not classified according to nuclide.
Locking Out	A control switch or valve handle which is fixed in either the open or closed position by one or more padlocks. The lock(s) may be removed only by designated supervisors.
Poison	A material having a high neutron capture cross section (e.g. boron, cadmium).

Sump	A small pit at the low point in the floor of processing areas which serves as a collection drain for liquids and equipped with means for detection and removal of accumulated liquids.
U-235 Equivalent Concentration	U-235 equivalent concentration (g/l) = 1.66 x Pu concentration (g/l) + 1.66 x U-233 concentration (g/l) + U-235 concentration (g/l). This formulation should not be applied to solution systems that depend on geometry control. The conversion analyses are based upon calculations reported in BNWL-3 and ORNL-TM-686.
Ready Condition	A temporary condition of operation using prescribed alternative instrumentation and controls or additional administrative safeguards while immediate corrective action is being taken in accordance with the requirements of the license.

4.0 SAFETY LIMITS

The limits established in this section of the Technical Specifications permit flexibility essential to chemical processing and define the boundaries of safe operations. The limits have been set above the values required by normal operations and below the values at which an accident could result or the safety of the public could be jeopardized. If specifications 4.1.1. and 4.2.1 are exceeded, an immediate shutdown of the operations leading to the occurrence is required pursuant to Section 7. Unlike a nuclear reactor which is designed to operate in a critical region, a chemical processing plant is designed and operated in such a way as to remain subcritical at all times. No single malfunction can lead rapidly to a critical incident; there is sufficient capacitance inherent in the process to ensure an extended time period before the limits for safe operation are exceeded. Secondary controls are installed to provide compensation in the event of the failure of a primary control. In addition, administrative controls are imposed to further ensure the safe operation of the facility. For these reasons, if the other specifications in this section are exceeded, shutdown is not mandatory. If such an occasion arises, plant operations may continue in a "ready condition" until normal operations are restored, however, if during this period there is any occurrence or indication of an occurrence that would further reduce the margin of safety an immediate shutdown is required.

4.1 RADIOACTIVITY CONTENT OF RELEASED GASEOUS EFFLUENTS

APPLICABILITY

This specification establishes limits for the discharge of Krypton-85 and Iodine-131 from the stack.

OBJECTIVE

To provide reasonable assurance that concentrations of radioactivity occurring off-site from the airborne release of radioactivity will meet with the requirements of Title 10, Code of Federal Regulations Part 20, and the guidance values established by the Federal Radiation Council.

SPECIFICATION

4.1.1 RADIOACTIVITY RELEASED FROM THE STACK SHALL BE LIMITED AS FOLLOWS:

a. KRYPTON-85: 1.26×10^4 CURIES PER DAY
b. IODINE-131 3.3 CURIES PER YEAR

4.1.2 PRIOR TO THE START OF A DISSOLUTION, METEOROLOGICAL DATA WILL BE EVALUATED (AND RETAINED) TO DETERMINE TO A HIGH DEGREE OF PROBABILITY THAT OFF-SITE CONCENTRATIONS OF KRYPTON-85 AND IODINE-131 WILL NOT EXCEED ACCEPTABLE LEVELS.

BASES

The discharge of radioactive effluents via the stack is discussed in paragraphs 7.6 through 7.9 of the Final Safety Analysis Report. It was shown therein that the only significant isotopes which will be discharged from the stack are Kr-85, I-129, I-131, Xe-131, Xe-133, and H-3. It was shown further that only krypton would be significant from a direct radiation dose standpoint, and that the control of Iodine-131 would afford adequate protection against a health hazard due to the ingestion of radioactive material. Particulate activity will not be released from the stack at concentrations in excess of the concentrations established by 10 CFR 20.106(a).

The Final Safety Analysis Report considered site meteorology in some detail. In paragraph 7.7 it was shown that on the basis of reasonable assumptions, the following dilutions would be applicable for diffusion from the stack exit to the site

boundary under the assumed normal and inversion meteorological conditions.

Normal (Slightly unstable): 5×10^6 cubic meters per second.

Inversion (Moderately stable): 1×10^5 cubic meters per second.

As stated in paragraph 2.12 to 2.14 of the Final Safety Analysis Report, the frequency of moderately stable conditions in this geographical area is expected to be low.

The Krypton-85 content of a design basis batch of nuclear fuel is listed as 6,300 curies in paragraph 7.6. The discharge of the Krypton-85 from two such batches per day would be diluted to an average concentration of 3×10^{-7} microcurie per cubic centimeter by a 4.36×10^5 cubic meters per second flow of air. This is less than one-tenth the normal aeolian dilution of 5×10^6 cubic meters per second expected in this area. Hence, the limitation of 12,600 curies per day Krypton-85 released from the stack provides reasonable assurance that acceptable concentrations will not be exceeded off-site.

In paragraph 7.6 the Iodine-131 content of a design basis batch of nuclear fuel is listed as 1.8 curies, and it is stated that the off-gas treatment equipment is designed to remove 99.5 per cent of the incident iodine so that only 9×10^{-3} curie would be discharged.

In the case of iodine, the Federal Radiation Council has suggested a radioactivity intake guide of 10^{-10} curies of I-131 per day by children. Assuming that a child will drink one liter of milk per day and that the radioactivity in a liter of milk is one-tenth that distributed over one square meter of grass, as observed in the Windscale incident, the permissible equilibrium concentration on the grass would be 10^{-9} curie per square meter of grass.

If 9×10^{-3} curie of Iodine-131 from a batch were discharged through the stack per day, a total of 3.3 curies per year, an average dilution with 1.0×10^6 cubic meters of air per second would be needed to maintain pasture concentrations of 10^{-9} curie per square meter, with deposition in equilibrium with radioactive decay.

$$\text{Dilution} = \frac{\text{Iodine Emission} \times \text{Deposition Velocity}}{\text{Decay Constant} \times \text{Pasture Concentration}}$$

$$= \frac{9 \times 10^{-3} \times 10^{-2}*}{8.6 \times 10^{-2} \times 10^{-9}}$$

$$= 1.0 \times 10^6 \text{ cubic meters per second}$$

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* Deposition velocity taken to be 0.01 meter per second from A. C. Chamberlin, Quarterly Journal of the Meteorological Society, Volume 85, No. 336, p.358.

This is about one-fifth of the normal aeolian dilution of 5×10^6 cubic meters per second expected in this area. Milk produced at the on-site farm and in neighboring pastures is to be tested routinely for Iodine-131 content to provide assurance that adequate dilution is afforded by the meteorological conditions that actually prevail. If the collected data show the calculation to be in error, the 3.3 curies per year Iodine-131 stack discharge limit will be subject to revision.

There are two safety factors not taken account of in the above calculation. Cattle will be on pasture approximately one-half of each year, and losses to the soil have been observed to reduce the effective half-life of Iodine-131 on grass to approximately 5 days. These factors would reduce the average dilution needed to 6.7 per cent of the 5×10^6 cubic meters per second normally expected value.

$$0.067 = \frac{5 \times 1.0 \times 10^6}{2 \times 8 \times 5 \times 10^6}$$

Abrupt complete failure of the iodine removal equipment is unlikely. But if this should happen coincident with stack monitoring failure and release all the Iodine-131 in a processing lot of fuel under inversion meteorological conditions, the maximum concentration in milk expected to result may be computed to be 1.8×10^{-8} curie per liter, and the projected dose to the thyroid of a child drinking 1 liter per day of milk following such an incident would be 2 rads which compares to the Federal Radiation Council Protective Action Guide of 30 rads to an individual.

4.2 RADIOACTIVITY CONTENT OF RELEASED LIQUID EFFLUENTS

APPLICABILITY

This specification establishes limits for the discharge of radioactivity contained in liquid effluents from the site.

OBJECTIVE

To provide reasonable assurance that concentrations of radioactivity in Cattaraugus Creek will meet the requirements of Title 10 Code of Federal Regulations Part 20.

SPECIFICATION

4.2.1 ACTIVITY RELEASED TO CATTARAUGUS CREEK FROM ALL SOURCES ON THE SITE SHALL NOT AT ANY TIME RESULT IN A CONCENTRATION, AS MEASURED AT THE MONITORING STATION IN CATTARAUGUS CREEK, WHICH EXCEEDS TWICE THE CONCENTRATION SPECIFIED IN TITLE 10 CODE OF FEDERAL REGULATIONS PART 20; PROVIDED HOWEVER, THAT IN ANY 365 CONSECUTIVE DAY PERIOD, THE AVERAGE CONCENTRATION OF RADIOACTIVITY IN CATTARAUGUS CREEK SHALL NOT EXCEED THAT SPECIFIED IN TITLE 10 CODE OF FEDERAL REGULATIONS PART 20. SUFFICIENT DATA ON STREAM FLOWS AND RADIOISOTOPE CONCENTRATIONS MUST BE RETAINED TO DEMONSTRATE COMPLIANCE WITH TITLE 10 CODE OF FEDERAL REGULATIONS PART 20.

BASES

This specification limits the radioactivity concentration in Cattaraugus Creek at any time to twice the annual average concentration permitted by Title 10 Code of Federal Regulations Part 20.106(a) on an arbitrarily conservative basis. The discharge of liquids from the site has been discussed in paragraph 7.11 through 7.13 of the Final Safety Analysis Report. The most significant isotope in the liquid effluent will be tritium for which a separate analysis will be made. If, after radioactive operations commence, it appears desirable to do so, separate analyses will also be made for other isotopes, such as Ru-106 and Zr-Nb-95, which combined activity shall be subtracted from the gross activity as per an alternative allowed by the notes to Appendix B of Part 20 of Title 10 of the Code of Federal Regulations. All other isotopes for which a specific analysis is not made will be determined on a gross count basis.

Continuous sampling will be conducted at the monitoring stations at the confluence of Frank's Creek and Quarry Creek and near the site boundary on Cattaraugus Creek. The composite sample from each station will be checked weekly. The flow in Cattaraugus Creek will be measured as necessary and discharge from the storage lagoons will be regulated to conform to the flow in Cattaraugus Creek.

Stream flow is measured with an instrument which records the level of the stream. The flow is calculated using the cross section of the stream bed and a series of velocity measurements at various water heights.

The consequences of exceeding this Technical Specification are that the limits of Title 10 Code of Federal Regulations Part 20 might, on a yearly basis, be exceeded in Cattaraugus Creek. However, unlike the discharge of gaseous activity--which once introduced into the ventilation stream will in all likelihood be released immediately--activity released into the liquid effluent system goes through 4 separate holdup units, viz, the interceptor, a 300,000 gallon lagoon, and two successive 2,000,000 gallon lagoons. Thus, there are four separate chances to apprehend errant activity and the total possible holdup time is five months. Plant operations, therefore, has a high degree of control over the discharge of radioactivity in the liquid effluent and there is plenty of time to shut the plant down in an orderly fashion without exceeding this Technical Specification.

4.3 FUEL QUANTITY AND CANISTER SPACING

APPLICABILITY

This specification establishes limits for the operation of the Cask Unloading and Fuel Storage Pools.

OBJECTIVE

To assure that individual units and arrays of units are maintained in subcritical configuration.

SPECIFICATION

4.3.1 IRRADIATED NUCLEAR FUEL IN THE FUEL STORAGE POOL SHALL BE STORED IN CANISTERS.

4.3.2 THE QUANTITY OF FUEL STORED IN A CANISTER SHALL BE LIMITED SO THAT THE EFFECTIVE NEUTRON MULTIPLICATION FACTOR(k_{eff}) SHALL NOT EXCEED 0.85 BASED ON COLD CLEAN FUEL. THE PRECISION OF THE k_{eff} CALCULATION SHALL BE CONFIRMED BY APPLYING THE CALCULATIONAL METHOD TO KNOWN CRITICAL SYSTEMS OF SIMILAR FUEL MATERIAL.

4.3.3 IN LIEU OF DETERMINING THE k_{eff} OF A CANISTER LOADING, ANY SINGLE FUEL ELEMENT OR PACKAGE WHICH HAS BEEN STORED UNDER WATER PRIOR TO SHIPMENT MAY BE STORED IN A CANISTER. FOR SOME OF THE PRESENTLY KNOWN FUELS, EXAMPLES OF ACCEPTABLE CANISTER LOADINGS ARE GIVEN BELOW IN LIEU OF k_{eff} CALCULATIONS.

Category I	Y-(1), 10 kg; D-(1), 2 kg; Pa-(1), 10 kg
Category II	(1), _____kg*
Category III	C-(1), 10 kg
Category IV	F-( ), _____kg*
Category V	Z-(1), 2.8 kg
Category VI	S-(24), 10 kg
Category VII	M-(8, 4 sets of 2 end to end), 2.6 kg
Category VIII	P-(1), 2 kg
Category IX	N-(full canister)

C--Con Ed Core A fuel
D--Dresden Fuel
F--Fermi Fuel (seed)
M--MTR Fuel (Up to 300 g U-235/element)
N--NPR Fuel
P--Piqua Fuel
Pa--Pathfinder Fuel
S--Scrup Nat. U Fuel
Y--Yankee Fuel
Z--PWR Fuel
( )--Number of Fuel Assemblies
kg--kilograms of U-235 in cold clean fuel
*--to be determined when fuels become available.

4.3.4 CANISTERS SHALL BE SPACED TO PROVIDE A MINIMUM OF 12 INCHES OF WATER BETWEEN FUEL CONTAINED IN ADJACENT CANISTERS EXCEPT FOR THOSE CANISTERS CONTAINING DRESDEN CANNED FUELS WHICH MAY BE SPACED TO PROVIDE 9.6 INCHES OF WATER BETWEEN ADJACENT FUEL CANS PROVIDED THAT THE IMMEDIATELY ADJACENT STORAGE RACK(S) REMAIN EMPTY.

BASES

The Fuel Receiving and Storage Area has been designed to permit the handling of fuel elements in such a manner as to provide that at all times they cannot react nuclearly with one another. Thus, to the largest extent possible, geometry backed up by administrative control is relied upon to prevent the interaction of one fuel element with another. These precautions are necessary since the FRS will at most times contain fuel in excess of that necessary to result in a critical assembly if placed in optimum array.

The storage rack and canisters provide a minimum edge to edge spacing of eight inches between adjacent canisters and a minimum edge to edge spacing of twelve inches between the fuel contained in adjacent canisters. A separation of twelve inches of water prevents significant interaction of neutrons and provides a safe array as indicated in "The Technology of Nuclear Reactor Safety" Volume I, p. 279.

A $k_{eff} \leq 0.85$ provides a reasonable margin of safety to account for uncertainty in single group calculations, some margin of error in the identification of the fissile material content, or operational error in canister loading. By comparing the calculative method with known critical systems of similar fuel material, uncertainties in the method are minimized.

Individual fuel elements or packages previously stored under water have been demonstrated as safe, and, therefore, no further calculations are deemed necessary provided that 12 inches is maintained between fuel in adjacent canisters in storage.

Administrative procedures are established to assure correct canister loadings.

The decreased water spacing in the case of Dresden Canned Fuel has been shown to be nuclearly safe. (Letter from NFS to USAEC Division of Materials Licensing dated April 28, 1965.)

The FRS has been designed to prevent the accumulation in critical array of sufficient fuel to cause a critical incident. The consequence of exceeding this Technical Specification would be to decrease the margin of safety for the prevention of criticality or, in final analysis, to result in a critical incident. In paragraphs 7.33 and 7.34 and 8.29 of the Safety Analysis it has been assumed

that despite all design efforts, a critical incident somehow does occur. The consequences of such an unlikely event have been analyzed to show that neither operating personnel nor the general public would be injured as a result thereof.

The action to be taken to correct an exceeding of this Technical Specification is to take immediate steps to increase spacing of fuel in storage or decrease canister loading since a storage pool cannot be "shut down" in the sense that an operating plant can.

4.4 DISSOLVER SOLUTION CONCENTRATION

APPLICABILITY

This specification establishes limits to govern the dissolver charging operation.

OBJECTIVE

To assure that solutions containing special nuclear material will remain at a subcritical concentration.

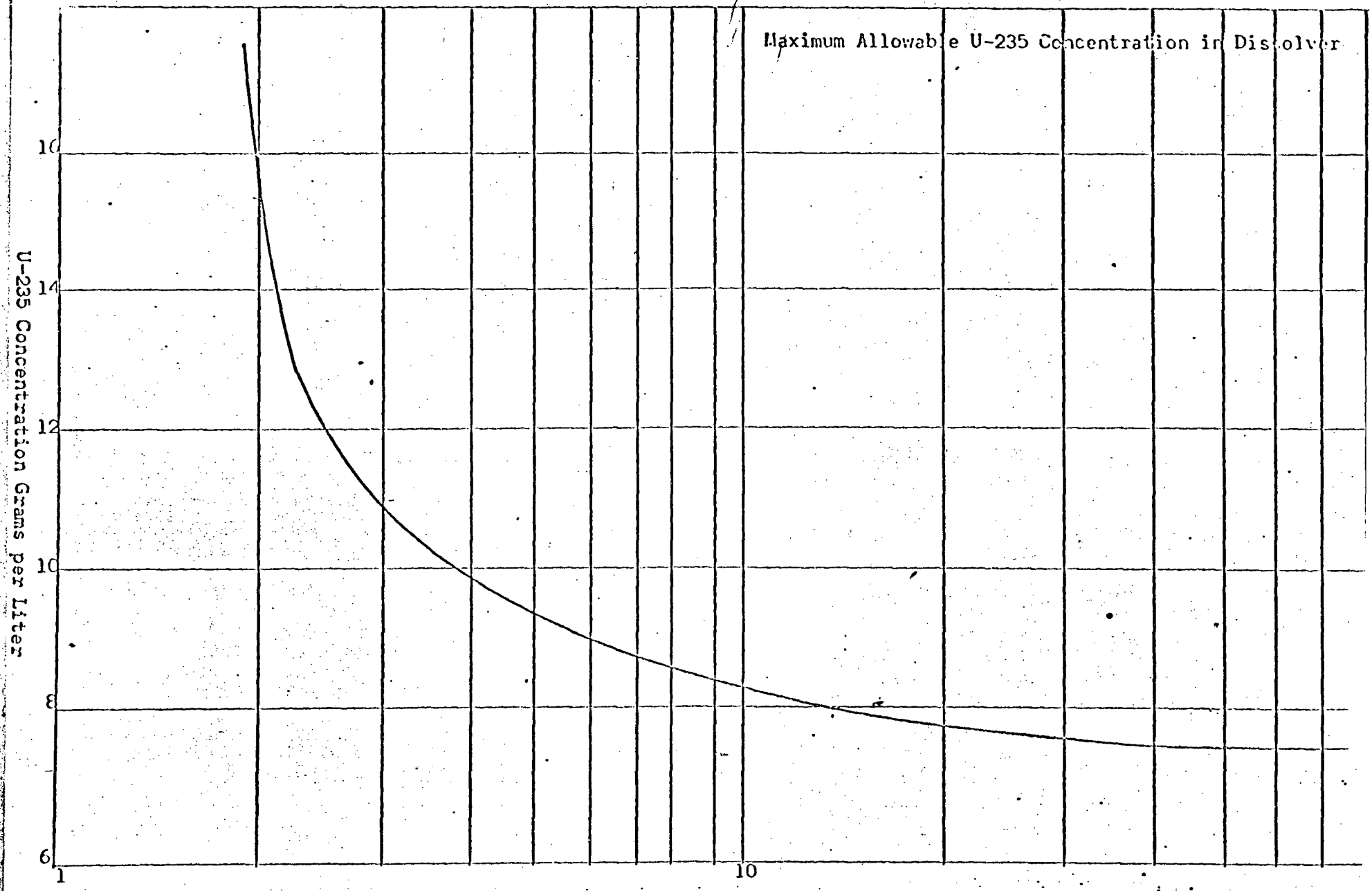
SPECIFICATION

4.4.1 THE TOTAL CHARGE OF FUEL AND DISSOLVENT SHALL BE CHOSEN SUCH THAT THE RESULTANT AVERAGE U-235 CONCENTRATION SHALL NOT EXCEED THE VALUES SHOWN IN THE FOLLOWING GRAPH. THE U-235 CONTENT OF THE CHARGED FUEL SHALL BE ASSUMED TO BE THE U-235 CONTENT OF THE COLD CLEAN FUEL.

BASES

In the dissolver and in the subsequent solvent extraction operations the primary control of criticality rests in the control of the concentration of fissionable material in the various solutions. This concentration is established initially in the dissolver by control of the charge of fuel and dissolvent to the dissolver by administrative procedures.

Although the lower portion of each dissolver is geometrically favorable, the enlarged upper portion is not. Since samples of the dissolver solution will not normally be taken, the limiting safe concentrations at various enrichments are based on the mass of U-235 in cold, clean fuel. The specified concentrations are 60% of the calculated critical concentrations reported in ORNL-TM-686, Limiting Critical Concentrations of Aqueous Nitrate Solutions of Fissile and Fertile Isotopes. The calculations in ORNL-TM-686, which were made with the IBM-7090 MODRIC neutron diffusion code, overestimate the experimentally determined critical concentration of fully enriched uranium by 3% and underestimate the experimentally determined critical concentration of 3.04% enriched uranium by 10%. These experimental determinations indicate that by setting the maximum concentration at 60% of the calculated critical concentrations there is an adequate margin of safety to provide for computational, analytical and gauging errors.



Weight Per Cent Enrichment U-235 in Aqueous Nitrate Solutions
(as cold clean fuel)
Figure 4.4.1

The consequence of exceeding this Technical Specification is to reduce the margin of safety in preventing criticality and in the ultimate sense would result in a critical incident. Although such an event would be detected by the plant monitoring system, no personnel injury would be received from neutron radiation from such an event since the dissolvers are shielded with 5'-9" of concrete. Gaseous activity might well be discharged from the stack in excess of Paragraph 4.1. This possibility has been analyzed in Paragraphs 7.30-7.32 and 8.24-8.28 of the Safety Analysis and it has been shown that even in the event such a critical incident were to occur there would not be injury to either plant personnel or the general public.

The action to be taken in the event of exceeding this Technical Specification is to make an immediate dilution of the solution to bring it within this specification.

4.5 FEED SOLUTION CONCENTRATION

APPLICABILITY

This specification establishes the concentration limits to be observed in the operation of the feed adjustmant and accountability tank.

OBJECTIVE

To assure that solutions containing special nuclear material will remain at a subcritical concentration.

SPECIFICATION

4.5.1 THE CONCENTRATION OF FISSIONABLE ISOTOPES IN THE FEED ADJUSTMENT AND ACCOUNTABILITY TANK SHALL NOT EXCEED THE U-235 EQUIVALENT CONCENTRATIONS SHOWN IN THE ACCOMPANYING CURVE.

BASES

The feed adjustment and accountability tank is not geometrically favorable, hence concentration control of the fissionable isotopic content of the tank must be maintained in order to ensure nuclear safety. The contents of this tank will be sampled to determine the actual fissionable isotopic concentrations. From this sample the U-235 equivalent concentration will be determined. The concentration may be increased by evaporation or decreased by dilution.

The specified concentrations defined by the accompanying curve are 60 per cent of the calculated critical concentrations as reported in ORNL-TM-686, Limiting Critical Concentrations of Aqueous Nitrate Solutions of Fissile and Fertile Isotopes. The calculations in ORNL-TM-686, which were made with the IBM 7090 MODRIC neutron diffusion code, overestimate the experimentally determined critical concentration of fully enriched uranium by 3 per cent and underestimate the experimentally determined critical concentration of 3.04 per cent enriched uranium by 10 per cent. These experimental determinations indicate that by setting the maximum concentration at 60% of the calculated critical concentrations there is an adequate margin of safety to provide for computational, analytical and gauging errors.

The consequence of exceeding this Technical Specification is to reduce the margin of safety in preventing criticality and in the ultimate sense would result in a critical incident. Although such an event would be detected by the plant monitoring system, no personnel injury would be received from neutron irradiation from such an event since the feed adjustment and accountability tank is

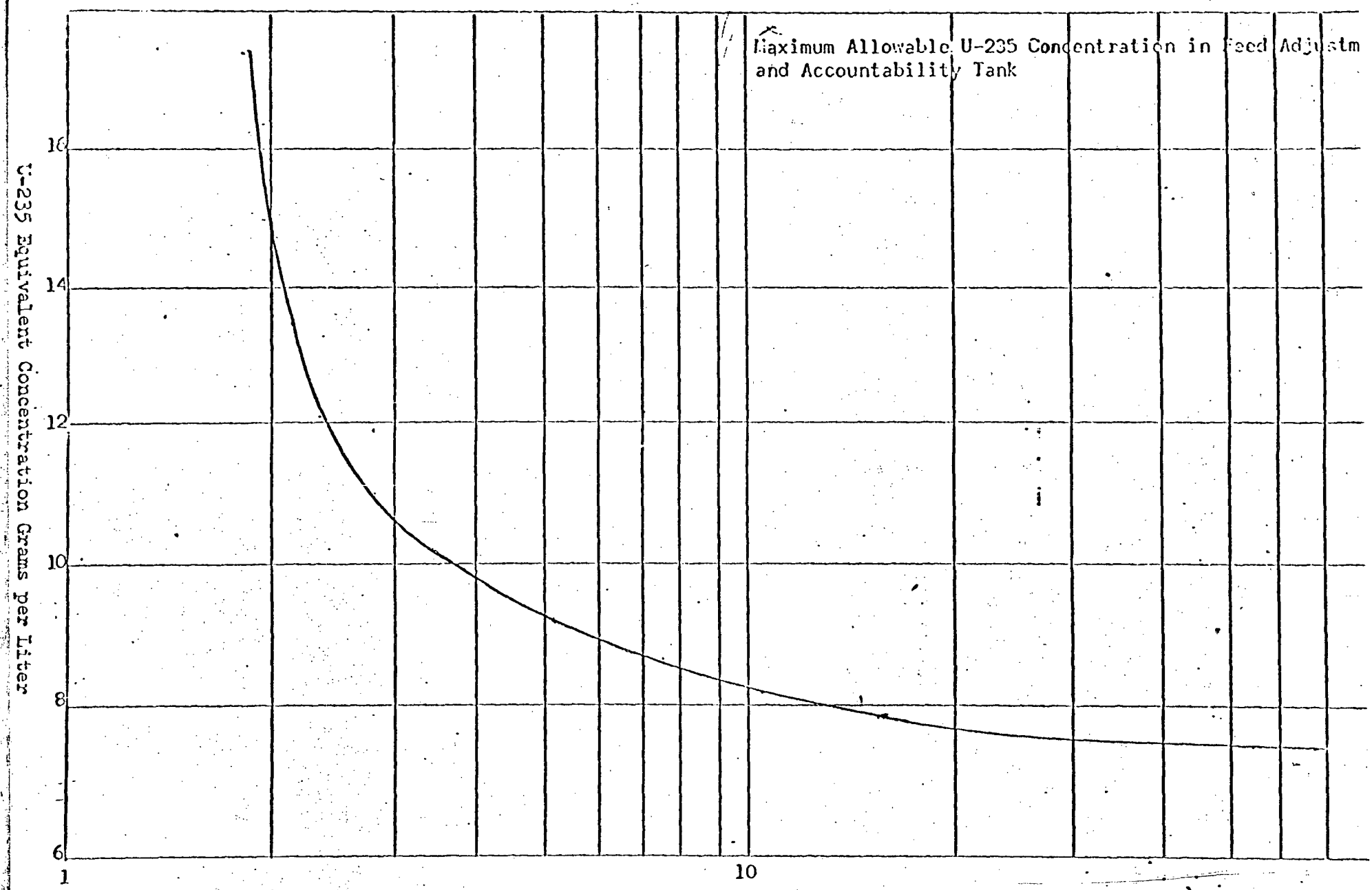

Maximum Allowable U-235 Concentration in Feed Adjustm
and Accountability Tank
U-235 Equivalent Concentration Grams per Liter
16
14
12
10
8
6
1
10
Weight Per Cent Enrichment U-235 in Aqueous Nitrate Solutions

Figure 4.5.1

shielded with 5'-9" of concrete. Gaseous activity might well be discharged from the stack in excess of Par. 4.1. This possibility has been analyzed in paragraphs 7.30-7.32 and 8.24-8.28 of the Safety Analysis and it has been shown that even in the event such a critical incident were to occur, there would not be injury to either plant personnel or the general public.

The action to be taken in the event of exceeding this Technical Specification is to make an immediate dilution of the solution to bring it within the Specification.

4.6 FISSIONABLE ISOTOPE CONCENTRATION IN SOLVENT EXTRACTION

APPLICABILITY

This specification establishes limits for the operation of the solvent extraction columns in the plant.

OBJECTIVE

To assure subcritical operation of the solvent extraction columns.

SPECIFICATION

4.6.1 EVEN UNDER CONDITIONS WHERE CONTROL OF ANY SINGLE VARIABLE IS LOST THE MAXIMUM CONCENTRATIONS OF FISSIONABLE ISOTOPES WITHIN THE COLUMNS SHALL NOT EXCEED VALUES CORRESPONDING TO A k_{eff} OF 0.95.

BASES

Throughout the solvent extraction system the primary control of criticality is provided by controlling the concentration of fissionable material. This Technical Specification is designed to provide a concentration of fissionable material less than that required to achieve criticality with a margin of safety of 0.05 k_{eff} units.

The flooding velocity of the limiting column determines the maximum flow of aqueous and organic through a series of columns. Table 4.21, as amended, of the Final Safety Analysis Report shows the nominal processing flow rates--the lower values, and the flooding rates--the higher values. The streams given in this table are numbered and are identified by this number on Drawing No. 15R-A-5.

Concentrations in the column terminal streams are controlled by: (1) limiting the fissionable isotope concentration in the feed (HAF) (see Spec. 4.5.1); (2) limiting the TBP content in the extractant to a value which cannot contain a critical concentration of fissionable isotopes within the solvent extraction columns, (see Spec. 4.7.1); and (3) maintaining stream ratios as shown in Table 4.21 as amended to $\pm$ 30% of the values shown.

The most critical factors for control of fissionable isotope concentration within the columns are, omission of nitric acid in the scrub streams on the extraction columns, or stoppage of the strip stream flow for appreciable periods of time on the stripping columns. Occurrence of these events under the most extreme conditions present no criticality hazards as shown by the data given in Table 6.143 of the Final Safety Analysis Report as amended. In determining the k_{eff} of 0.95 for the partition and uranium solvent extraction columns which contain U-233 or Pu, the reactivity based upon the analytically determined fissionable isotope concentrations will be taken into account.

The consequence of exceeding this Technical Specification is to reduce the margin of safety in preventing criticality and in the ultimate sense would result in a critical incident. Although such an event would be detected by the plant monitoring system, no personnel exposure in excess of the guidelines of Title 10 Code of Federal Regulations Part 100 would be received from neutron irradiation from such an event since no part of the solvent extraction equipment is shielded with less than 3 feet of concrete. Gaseous activity might well be discharged from the stack in excess of Paragraph 4.1. This possibility has been analyzed in paragraphs 7.30-7.32 and 8.24-8.28 of the Safety Analysis and it has been shown that even in the event such a critical incident were to occur there would not be injury to either plant personnel or the general public.

If this specification is found to be exceeded and if immediate operating steps cannot be taken to correct the fact, the solvent extraction system must be shut down until the error can be corrected.

4.7 EXTRACTANT CONCENTRATION

APPLICABILITY

This specification establishes the limits for extractant concentrations which may be used in the extraction process for various fuel enrichments.

OBJECTIVE

To limit the concentrations of fissionable isotopes in the solvent during column operations to a subcritical concentration.

SPECIFICATION

4.7.1 CONCENTRATION OF THE EXTRACTANT FOR THE VARIOUS FUEL CATEGORIES LISTED IN SECTION 2.2 SHALL NOT EXCEED THE FOLLOWING:

MAXIMUM COLD CLEAN FUEL ENRICHMENT WT. % U-235	EXTRACTANT CONCENTRATION VOL PER CENT TBP
10	33
26.5	11
100	11*
100	6

BASES

The distribution coefficient, and hence the concentration of uranium in the extractant stream, is a function of, among other things, the solvent (TBP) concentration in the extractant stream, i.e., the higher the TBP concentration, the higher the uranium concentration at equilibrium. In the event malfunction or maloperation of the equipment results in stoppage of extractant flow in the column, the concentration or uranium in the organic will build up to a level limited by the TBP concentration in the extractant. To ensure that these levels are well within safe operation, this specification limits the TBP concentrations as noted. Details

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* Processing of Zr-U alloy fuels requires a higher concentration of TBP in the partition cycle to obtain distribution ratios equivalent to those obtained with 5% TBP for other alloy fuels. Consequently a solvent TBP content not exceeding 11 v/o will be used on the partition cycle. The remaining extraction cycles will use 5 v/o TBP. Due to the lower uranium distribution ratios in the presence of fluoride, the solvent phase can contain no more uranium with 11% TBP than it can with 5% TBP in the absence of the fluoride. Consequently the hazards are no greater with the higher TBP content than with 5% for this system in the partition cycle.

of these limits are discussed in the Final Safety Analysis Report in Paragraphs 6.139 to 6.146 and as indicated therein, the solvent concentration limits specified provide a factor of safety of approximately three.

The consequence of exceeding this Technical Specification is to reduce the margin of safety in preventing criticality and in the ultimate sense could lead to a critical incident. Although such an event would be detected by the plant monitoring system, no personnel exposure in excess of the guidelines of Title 10 Code of Federal Regulations Part 100 would be received from neutron irradiation from such an event since no part of the solvent extraction equipment is shielded with less than 3 feet of concrete. Gaseous activity might well be discharged in excess of Paragraph 4.1. This possibility has been analyzed in Paragraphs 7.30-7.32 and 8.24-8.28 of the Safety Analysis and it has been shown that even if such a critical incident were to occur, there would be no injury to either plant personnel or the general public.

If this Technical Specification is found to be exceeded and if immediate operating steps cannot be taken to correct the fact, the solvent extraction system must be shut down until the error can be corrected.

4.8 URANIUM PRODUCT SOLUTION STORAGE: CONCENTRATION LIMITS

APPLICABILITY

This specification establishes the concentration limits for the storage of uranium product solutions.

OBJECTIVE

To assure that the uranium product solutions will be stored under nuclearly safe conditions.

SPECIFICATION

4.8.1 URANYL NITRATE SOLUTIONS SHALL NOT BE STORED EXCEPT AS FOLLOWS:

4.8.1.1 URANYL NITRATE SOLUTION MAY BE STORED IN ANY VESSEL IF THE U-235 CONCENTRATION IS NO GREATER THAN TABULATED BELOW AND PROVIDED THE U-233 PLUS PU CONTENT IS NOT GREATER THAN 1% OF THE U-235 CONTENT.

MAXIMUM ENRICHMENT WT. % U-235 IN U	U-235 CONCENTRATION GRAMS PER LITER
2.0	21.6
3.0	15.0
4.0	13.3
5.0	12.5
6.0	12.0
7.0	11.7
8.0	11.5
9.0	11.3
10.0	11.2
100.0	10.0

4.8.1.2 URANYL NITRATE SOLUTION CONTAINING U-233 MAY BE STORED IN ANY VESSEL IF THE COMBINED U-233 AND U-235 CONCENTRATION DOES NOT EXCEED 10.0 GRAMS PER LITER AND PROVIDED THAT THE PU CONTENT IS NOT GREATER THAN 1% OF THE COMBINED U-233 AND U-235 CONTENT.

4.8.1.3 URANYL NITRATE SOLUTION MAY BE STORED IN FIXED POISON TANKS 5D-13A, B AND C AT A CONCENTRATION NOT EXCEEDING 360 GRAMS U-235 PER LITER PROVIDED THE U-233 AND PU CONTENT IS NOT GREATER THAN 1% OF THE U-235 CONTENT.

4.8.1.4 URANYL NITRATE SOLUTION CONTAINING U-233 MAY BE STORED IN FIXED POISON TANKS 5D-13 A, B AND C IF THE CONCENTRATION OF U-233 AND U-235 COMBINED DOES NOT EXCEED 250 GRAMS PER LITER, PROVIDED (a) THE U-233 CONTENT IS NOT GREATER THAN 10% OF THE COMBINED U-233 AND U-235 CONTENT AND (b) THE PU CONTENT IS NOT GREATER THAN 1% OF THE COMBINED U-233 AND U-235 CONTENT.

4.8.1.5 SOLUTIONS STORED IN ACCORDANCE WITH 4.8.1.3 AND 4.8.1.4 ABOVE MAY BE PACKAGED IN CONTAINERS WHICH SHALL BE PLACED IN A RIGID STAINLESS STEEL CONTAINER WITH AN INSIDE DIAMETER NOT EXCEEDING 5.1 INCHES AND A LENGTH NOT EXCEEDING 53 INCHES. THIS STAINLESS STEEL CONTAINER SHALL BE SECURELY CENTERED AND TOTALLY ENCLOSED IN A RIGID STRUCTURE, THE MINIMUM OUTER DIMENSIONS OF WHICH ARE AT LEAST 22.5 INCHES. UNITS SO PACKAGED SHALL BE STORED IN THE BIRDCAGE STORAGE AREA AND SHALL NOT BE STORED ONE ON TOP OF ANOTHER.

BASES

The primary control of criticality in the storage of uranium product solutions is concentration of fissionable isotopes. This concentration control is backed up with fixed poisons (in case of tanks 5C-13A, B, and C) and geometry in the case of birdcage type shipping containers.

The limiting concentrations for U-235 specified in 4.8.1.1 are 80% of the calculated critical concentrations reported in ORNL-TM-686. The specified concentrations defined by the accompanying curve are 80 per cent of the calculated critical concentrations as reported in ORNL-TM-686, Limiting Critical Concentrations of Aqueous Nitrate Solutions of Fissile and Fertile Isotopes. The calculations in ORNL-TM-686, which were made with the IBM 7090 MODRIC neutron diffusion code, overestimate the experimentally determined critical concentration of fully enriched uranium by 3 per cent and underestimate the experimentally determined critical concentration of 3.04 per cent enriched uranium by 10 per cent. An 80% of calculated critical concentrations is justifiable for such operations because there are product solutions and the analysis of the constituents is known within an accuracy of ± 3 per cent.

The limiting concentration of combined U-233 and U-235 specified in 4.8.1.2 is based on the recommendations contained in TID-7016, Revision 1, Table I.

The limiting concentrations of U-235 specified in 4.8.1.3 and combined U-233 and U-235 specified in 4.8.1.4 are based on the recommendations of Work Group 3 of Subcommittee 8 of the Standards Committee of the American Nuclear Society. The concentration limit in 4.8.1.4 also takes into account subsequent storage in the containers specified in 4.8.1.5.

Provisions are made for the planar storage of 39 such birdcages in the Product Packaging and Shipping Area. Data presented in RFP-437 indicate a 5 x 477 array as being required for a nuclear configuration. By restricting storage to a planar array the degree of safety is about a factor of 60.

The concentrations of uranium in the product solutions covered by this Technical Specification are known more accurately than at any other point in the process. The very high unit value of these materials requires that the closest possible analytical control be maintained as a part of the accountability program. Replicate samples will be taken and analyzed at each of the points covered and a cross check will exist because at this stage material balances will be run to assure that process guarantees are met.

The consequences of exceeding any of these concentrations would be to lower the margin of safety discussed herein and in the ultimate sense would result in a critical incident. Although such an event would be detected by the plant monitoring system, no personnel exposure in excess of the guidelines of Title 10 Code of Federal Regulations Part 100 would be received from neutron irradiation from such an event for those vessels inside the cells since this equipment is shielded with 3 feet of concrete. Gaseous activity might well be discharged from the stack in excess of Paragraph 4.1. This possibility has been analyzed in paragraphs 7.10-7.32 and 8.24-8.28 of the Safety Analysis and it has been shown that even in the event such a critical incident were to occur the guidelines of Title 10 Code of Federal Regulations Part 100 would not be exceeded.

When the final product solution is brought out for shipping, it is no longer shielded. Hence, a critical incident at this point could do damage to plant personnel (but not the general public) from direct neutron irradiation. Therefore, close administrative controls are maintained to assure that concentrations are known and double-checked before product solution is brought out from the shielding. It should be noted that if the concentration were such as to cause a critical incident at this point, it would already have gone critical in the storage tank. The high enriched system is further protected by the fact that geometric control is used throughout the measurement and shipping operations.

If these specifications are found to be exceeded, no further fissionable material may be added to this system until the concentration has been corrected which correction must be carried out immediately.

4.9 PLUTONIUM ION EXCHANGE OPERATION

APPLICABILITY

This specification establishes limits for the operation of the ion exchange columns.

OBJECTIVE

To assure that a fire or explosion will not occur in the plutonium ion exchange columns.

SPECIFICATION

4.9.1 PLUTONIUM NITRATE SOLUTIONS SHALL NOT BE MAINTAINED IN CONTACT WITH ION EXCHANGE RESINS FOR MORE THAN 24 HOURS IN A STATIC CONDITION PRIOR TO ELUTION. THE PLUTONIUM NITRATE SOLUTIONS IN THIS SYSTEM SHALL AT ALL TIMES BE KEPT BELOW A MAXIMUM TEMPERATURE OF 80 DEGREES CENTIGRADE, AND BELOW A NITRIC ACID CONCENTRATION OF 8M.

BASES

Because of several incidents at chemical processing plants where plutonium was being purified or otherwise treated by ion exchange technique, the Hanford Laboratories and others investigated the variables which must be controlled to prevent explosions or fires caused by the autoignition of resins. Although autoignition of ion exchange resins is possible without plutonium, the severity of the reaction is increased to a marked degree by the presence of plutonium.

The Hanford work showed that a threshold temperature of about 115°C (239°F) is necessary for initiation of runaway self-heating which can result in a violent deflagration. Thus a margin of safety of 35°C is provided.

The temperature is maintained by heating water in an external heat exchanger to the control temperature. This water is then circulated through a coil surrounding the ion exchangers. Work with other variables has not been as well defined, but it is recommended in Reactor Fuel Processing, Vol. 7, No. 4, Fall 1964, (pp. 303 and 304) that the concentration of nitric acid used be no greater than 8M and that the resin-acid contact time should not exceed 24 hours. Administrative controls are imposed to insure these limits are not exceeded.

The consequence of exceeding this specification is to decrease the margin of safety discussed herein and in the ultimate sense could lead to autoignition of the resins. The worst consequence thereof would be to rupture the resin columns spreading plutonium solution through the PPC and perhaps through parts of the ventilation system. This would require a costly and difficult cleanup but would not result in exposure of the general public. Cleanup could be accomplished without harm to the operating personnel.

If this specification is found to be exceeded the condition must be corrected at once and no further plutonium may be loaded onto the column until the situation has been remedied.

4.10 PLUTONIUM PRODUCT SOLUTION STORAGE

APPLICABILITY

This specification establishes the limits for the storage of Plutonium Product Solutions.

OBJECTIVE

To assure that Plutonium Product Solutions are stored in a nuclearly safe manner.

SPECIFICATION

4.10.1 PLUTONIUM NITRATE PRODUCT SOLUTIONS SHALL NOT BE STORED EXCEPT AS FOLLOWS:

4.10.1.1 PLUTONIUM NITRATE SOLUTION MAY BE STORED IN NUCLEARLY SAFE TANKS 5D-5A, 5D-5B, and 5D-17.

4.10.1.2 PLUTONIUM NITRATE SOLUTION MAY BE PACKAGED IN CONTAINERS WHICH SHALL BE PLACED IN A RIGID STAINLESS STEEL CONTAINER WITH AN INSIDE DIAMETER NOT EXCEEDING 5.1 INCHES AND A LENGTH NOT EXCEEDING 53 INCHES. THIS STAINLESS STEEL CONTAINER SHALL BE SECURELY CENTERED AND TOTALLY ENCLOSED IN A RIGID STRUCTURE, THE MINIMUM OUTER DIMENSIONS OF WHICH ARE AT LEAST 22.5 INCHES. UNITS SO PACKAGED SHALL BE STORED IN THE BIRDCAGE STORAGE AREA AND SHALL NOT BE STORED ONE ON TOP OF ANOTHER.

BASES

The primary control of criticality in handling plutonium product solutions is geometry. This specification provides that only geometrically safe systems may be used for the storage of plutonium solutions. Slab tanks 5D-5A and 5D-5B are 2-1/2 inches thick and are provided with boral reflectors. Plutonium product is collected in these nuclearly safe (at all concentrations) slab tanks at about 200 g/l (Final Safety Analysis Report, Paragraph 6.159). Plutonium is transferred to birdcages from a 4" diameter measuring tank. The birdcages contain a centered 5 inch ss pipe not over 53 inches long; when fully reflected these systems are nuclearly safe to 500 g Pu/l as shown in TID-7016 Rev. 1 Fig. 7.

Provision is made for the planar storage of 39 such birdcages in the Product Packaging and Shipping area which limitation leaves a large margin of safety.

The consequence of using anything other than geometrically safe systems for the handling of plutonium solutions is to reduce markedly the margin of safety (which is essentially infinite in geometrically safe systems) and to provide the possibility of a critical incident in an unshielded area. Nothing but geometrically safe systems are provided for handling plutonium solutions and there should be no occasion to violate this specification. If it is found to be exceeded nothing further must be done in the plutonium packaging area until the condition has been corrected under close administrative control.

4.11 REWORK SOLUTION CONCENTRATION

APPLICABILITY

This specification establishes concentration limits to be observed in operations involving the Rework Evaporator and the Rework Evaporator Feed Tank.

OBJECTIVE

To assure that the solution containing special nuclear material will remain subcritical in both the Rework Evaporator and the Rework Evaporator Feed Tank.

SPECIFICATION

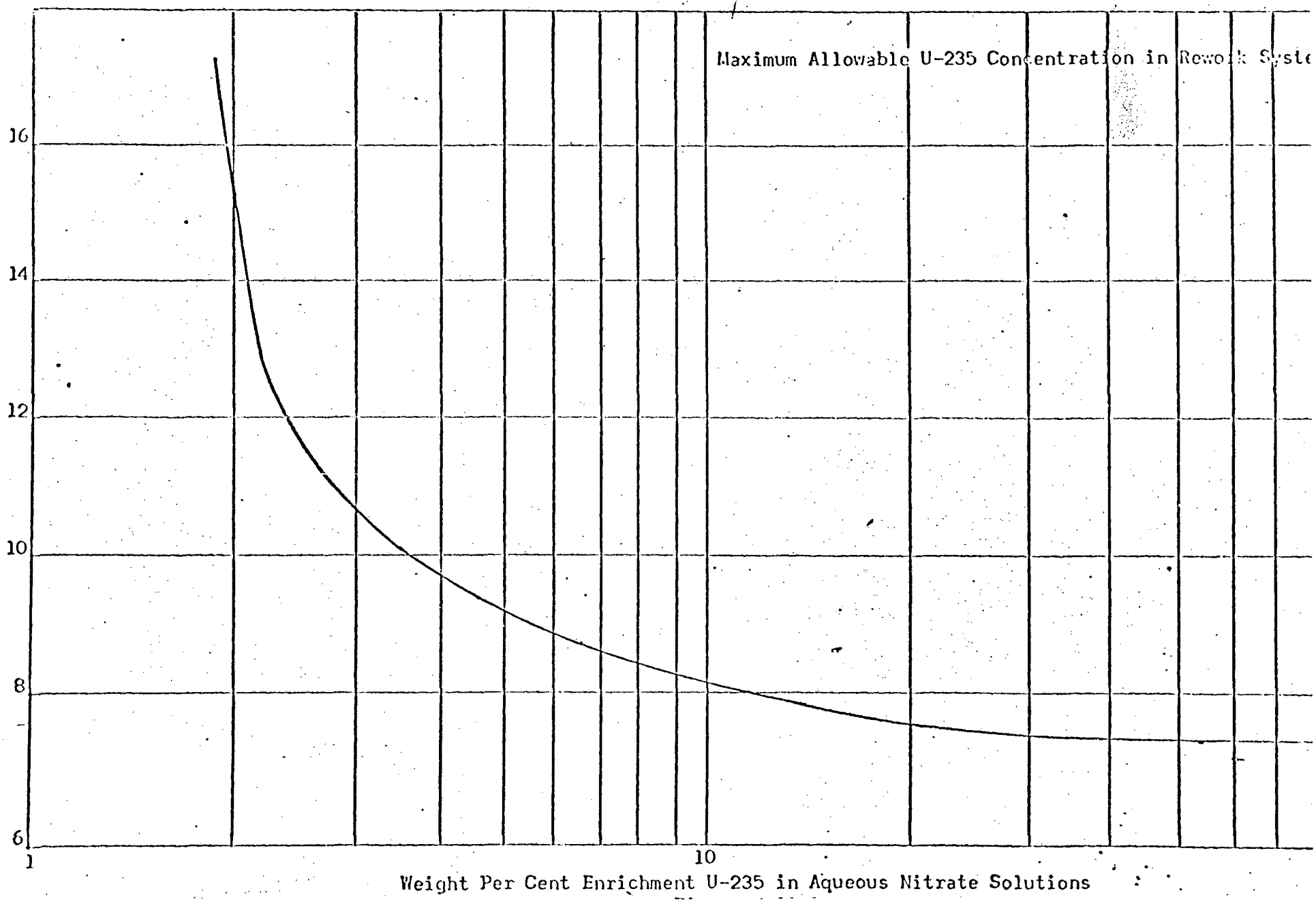
4.11.1 THE CONCENTRATION OF FISSIONABLE ISOTOPES IN THE REWORK EVAPORATOR AND THE REWORK EVAPORATOR FEED TANK SHALL NOT EXCEED THE U-235 EQUIVALENT CONCENTRATIONS SHOWN IN THE ACCOMPANYING CURVE.

BASES

The Rework Evaporator is not geometrically favorable hence concentration control of the fissionable isotopic content of the tank must be maintained in order to ensure nuclear safety. Any solutions entering the rework system will be sampled to determine the actual fissionable isotope concentrations. From this sample the U-235 equivalent concentration will be determined.

The safe concentrations defined by the accompanying curve are 60 per cent of the calculated critical concentrations as recommended and reported in ORNL-TM-686, Limiting Critical Concentrations of Aqueous Nitrate Solutions of Fissile and Fertile Isotopes. The calculations in ORNL-TM-686, which were made with the IBM 7090 MODRIC neutron diffusion code, overestimate the experimentally determined critical concentration of fully enriched uranium by 3 per cent and underestimate the experimentally determined critical concentration of 3.04 per cent enriched uranium by 10 per cent. These experimental determinations indicate that by setting the maximum concentration at 60% of the calculated critical concentrations there is an adequate margin of safety to provide for computational, analytical and gauging errors.

The consequence of exceeding the concentrations covered by this specification is to reduce the margin of safety discussed herein and in the ultimate sense could result in a critical incident. Although such an event would be detected by the plant monitoring system, no personnel exposure would be received from neutron irradiation from such an event since the vessels covered are


Maximum Allowable U-235 Concentration in Rework Syste
16
14
12
10
8
6
U-235 Equivalent Concentration Grams per Liter
1
10
Weight Per Cent Enrichment U-235 in Aqueous Nitrate Solutions

shielded by 5'-9" of concrete. Gaseous activity might well be discharged from the stack in excess of Paragraph 4.1. This possibility has been analyzed in paragraphs 7.30-7.32 and 8.24-8.28 of the Safety Analysis and it has been shown that even in the event such a critical incident were to occur, there would not be injury to either plant personnel or the general public.

If this specification is found to be exceeded, no further fissionable material will be added to the rework system until the situation is corrected; and the remedial action must be taken immediately.

4.12 WASTE STORAGE TANK CAUSTIC CONCENTRATION

APPLICABILITY

This specification establishes limits for the minimum concentration of caustic in Waste Storage Tanks, 8D-1 and 8D-2.

OBJECTIVE

To assure that the liquid waste in the waste storage tanks (8D-1 and 8D-2) shall be maintained in alkaline condition at all times.

SPECIFICATION

4.12.1 EXCESS CAUSTIC (BASED ON THE STOICHIOMETRIC AMOUNT REQUIRED TO NEUTRALIZE) IN THE HIGH LEVEL WASTE STORAGE TANKS 8D-1 AND 8D-2 SHALL BE PRESENT IN CONCENTRATION OF (i) AT LEAST 10% BASED ON THE WASTE VOLUME INTRODUCED THEREIN UNTIL THE TOTAL VOLUME OF INTRODUCED WASTE EXCEEDS 10,000 GALLONS; (ii) AT LEAST 5% ON THE SAME BASIS UNTIL THE TOTAL VOLUME OF INTRODUCED WASTE EXCEEDS 100,000 GALLONS; AND (iii) AT LEAST 1% ON THE SAME BASIS AFTER THE VOLUME OF INTRODUCED WASTE EXCEEDS 100,000 GALLONS.

BASES

In order to minimize the corrosion rate of the carbon steel high level liquid waste storage tanks 8D-1 and 8D-2, it is necessary to neutralize the normally acidic liquid waste. The procedure calls for maintaining excess caustic in the waste storage tank and is as follows: The empty waste storage tanks will be initially filled 1/3 to 1/2 full of water to which will be added sufficient caustic such that the addition of neutralized waste with a 1% excess caustic will result in at least a 10% excess caustic in the waste storage tank for the first 10,000 gallons of waste and at least a 5% excess caustic for the next 90,000 gallons of waste. One per cent excess caustic will be maintained in the tank for waste solution additions above 100,000 gallons. The reason for adding excess caustic is to prevent corrosion resulting from any inadvertent acidic liquid waste additions.

This amount of caustic will result in a pH of about 13 which is much higher than has been used in this type of storage system in the past. The amount of caustic could be reduced considerably still leaving an excellent margin of safety.

There is no consequence at all of not meeting this specification so long as the pH of the stored solution remains above 7. If the pH of the solution falls below 7 a rapid increase of the tank corrosion rate would result and failure of the tank could be expected. If this were to happen, a very serious financial loss would occur but, as discussed in paragraphs 7.24-7.29 of the Safety Analysis, no damage would occur to the general public.

4.13 SOLID RADIOACTIVE WASTE BURIAL

APPLICABILITY

This specification establishes the limits to be observed in operations relating to solid radioactive waste burial.

OBJECTIVE

To assure retention of activity from buried wastes.

SPECIFICATION

4.13.1 RADIOACTIVE SOLID WASTES GENERATED BY THE OPERATION OF THE FACILITY SHALL BE BURIED COMPLETELY WITHIN SILTY TILL. BEFORE A NEW TRENCH IS USED, IT WILL BE INSPECTED TO ASSURE THAT IT IS FULLY WITHIN THE SILTY TILL FORMATION. BURIAL SHALL BE RESTRICTED TO THE PLANT AREA DRAINED BY QUARRY CREEK AND ERDMAN BROOK. NO BURIAL OF WASTES SHALL BE NEARER THAN 100 FEET TO THE CREST OF THE DEFILES IN WHICH QUARRY CREEK AND ERDMAN BROOK FLOW. THE MINIMUM COVER OF SILTY TILL OVER THE WASTES SHALL BE FOUR FEET. SUITABLE MEANS SHALL BE PROVIDED AND MAINTAINED TO MINIMIZE EROSION IN THE SUBJECT DEFILES, BETWEEN THESE DEFILES AND BURIAL AREAS, AND OF THE EARTH COVER AFTER THE EXCAVATIONS HAVE BEEN FINALLY BACKFILLED.

4.13.2 THE LOCATION AT WHICH RADIOACTIVE SOLID WASTES ARE BURIED IN ACCORDANCE WITH SPECIFICATION 4.13.1 SHALL BE MARKED WITH CONCRETE CAIRNS. A PLOT SHOWING THE APPROXIMATE LOCATION OF ALL WASTES BURIED SHALL BE MAINTAINED ACCOMPANIED BY AN INDEX DESCRIBING THE GENERAL TYPES OF WASTES BURIED AT EACH LOCATION INCLUDING THE DATES OF BURIAL AND CLOSURE. TO PREVENT THEIR LOSS, DUPLICATE RECORDS OF BURIALS SHALL BE MAINTAINED IN SEPARATE LOCATIONS.

4.13.3 SOLID RADIOACTIVE WASTE TO BE BURIED SHALL BE PLACED IN PACKAGES WHICH SHALL PREVENT DISPERSION OF CONTENTS AND PREVENT CONTAMINATION OF HANDLERS. IF A PACKAGE IS RUPTURED WHEN PLACED IN THE TRENCH, EARTH OVERFILL SHALL BE IMMEDIATELY PLACED OVER THE RUPTURED PACKAGE.

BASES

This Technical Specification is established to assure that wastes are buried in such a manner as to assure that the radioactivity so buried will remain within the confines of the burial area within all established discharge limits.

Paragraph 2.45 of the Final Safety Analysis Report points out that the location specified for burial is of such a favorable geological formation that any activity in the ground water would eventually appear in the spanning streams, which are subject to monitoring, and nowhere else

As discussed in paragraphs 4.90, 7.14, 7.15, 7.16, and 7.17 of the Final Safety Analysis Report, the ion exchange and permeability data for the soil in which the activity is to be buried indicate that the

activity will be retained in the immediate vicinity of the source. Further, the low permeability of the silty till will delay any possible seepage so greatly that the longest lived ruthenium, an element which is not bound as well by ion exchange properties of the soil as the other significant nuclides, would undergo nearly complete radioactive decay before it could traverse 100 feet of this medium to an adjacent water course.

The provision of markers and records of burial on state owned property are to facilitate perpetual care and to preclude inadvertent excavation of radioactive material.

Exceeding this specification would merely increase the chance that radioactivity might escape the burial area. But the chance that the rate of escape would exceed the established discharge limits is very small.

5.0 MINIMUM OPERABLE EQUIPMENT REQUIREMENTS

The specifications included in this section are those that define the minimum amount of equipment necessary to safety which must be operable upon the initiation of processing operations and the alternates which may be used if this equipment should fail during processing operation. Those operations which are permitted with an alternate shall be considered to be continuing under "ready conditions". Unless otherwise specified, if the original equipment and the specified alternate fail during an operation involving the equipment, the operation shall be shutdown except in the case of the waste tank off-gas equipment, until repairs of both original and alternate equipment are accomplished.

5.1 EFFLUENT MONITORING

APPLICABILITY

This specification establishes the instrumentation requirements for monitoring the radioactive content of plant effluents.

OBJECTIVE

To assure that monitoring equipment will be in operation to permit appraisal of conformance with technical specifications governing the discharge rate of liquid and gaseous radioactive effluents

SPECIFICATION

5.1.1 A CONTINUOUS SAMPLER WHICH OBTAINS A REPRESENTATIVE SAMPLE OF CATTARAUGUS CREEK IMMEDIATELY BELOW THE WESTERN NEW YORK NUCLEAR SERVICE CENTER SHALL BE IN OPERATION WHEN EVER LIQUID IS RELEASED FROM THE STORAGE LAGOONS. IF THE SAMPLER FAILS, LAGOON DISCHARGES ARE PERMITTED FOR A PERIOD NOT EXCEEDING 14 DAYS, PROVIDED THAT SAMPLES OF CATTARAUGUS CREEK ARE COLLECTED MANUALLY WITH SUFFICIENT FREQUENCY TO VERIFY CONFORMANCE WITH SPECIFICATION 4.2.1 AND PROVIDED THAT IMMEDIATE AND CONTINUED ACTION SHALL BE TAKEN TO RETURN THE SAMPLER TO SERVICE.

5.1.2 A STACK MONITORING SYSTEM, WHICH RELIABLY MEASURES THE RADIOACTIVITY FOR WHICH LIMITS ARE ESTABLISHED IN SPECIFICATION 4.1.1 MUST BE IN OPERATION: 1. TO DETECT NOBLE GASES (SPECIFICALLY KRYPTON-85) OR IODINE DURING THE FIRST HOUR OF ANY DISSOLUTION OPERATION IN THE CPC OR ANY IRRADIATED FUEL CHOPPING OPERATION IN THE PMC; 2. TO DETECT PARTICULATE RADIOACTIVITY DURING ANY CHOPPING OPERATIONS IN THE PMC OR PRODUCT LOADOUT OPERATIONS.

BASES

Technical Specifications 4.1 and 4.2 represent the basic working limits at which radioactivity may be discharged from the plant into the atmosphere and watercourses respectively. The determination that these limits have been met is made by using air monitors in the stack and a sampler in Cattaraugus Creek. Because of the importance of these measurements, it is reasonable to establish specifications for the availability of this monitoring and sampling equipment.

Generally the storage lagoons have sufficient holdup capacity to permit storage of plant effluent if the continuous sampler in Cattaraugus Creek becomes inoperative. However, it is possible that a failure of the sampler could occur during the period of time during which it would be desirable to continue to release material stored in the lagoons. To allow operation to continue during such a period, manual sampling may be substituted for a maximum of 14 days during which repair of the sampler could be accomplished. The frequency of manual sampling will be determined by the flow rate in Cattaraugus Creek, the discharge rate from the lagoon and activity content of the lagoon in order to assure that the limits in Specification 4.2.1 are met.

Two monitors, each of which monitors for Kr-85, I, and particulate (described in NFS letters to the USAEC Division of Material Licensing dated January 25, 1965, p. 6 and February 17, 1965, pp. 3-5) have been provided to monitor radioactive stack effluents. There is, therefore, a high degree of probability that at least one monitor will be available throughout mechanical and chemical processing operations. Both units will be available at the start of campaign operations, however, even if both units should fail after the initiation of processes involving the evolution of noble gases, iodine, or particulate, the amount of activity which could be released would not represent a serious risk to the health of the public.

A specific example follows: If the monitor failures were accompanied by an iodine removal system failure which permitted the release of the total radioiodine content in one batch of fuel during a period of unfavorable meteorological condition (inversion), the maximum radioiodine concentration in milk off-site would not be likely to exceed 1.8×10^{-8} curies per liter, and the projected dose to the thyroid of a child drinking 1 liter per day of milk following such an incident would be 2 rads which compares to the Federal Radiation Council Protective Action Guide of 30 rads to an individual (Ref. Final Safety Analysis Report paragraphs 7.6 - 7.9).

Additional supporting statements are contained in the Final Safety Analysis Report paragraphs 6.10 and 7.6. It is concluded that because of the filtration of off gas from the PMC, where it is expected that the greatest amount of dry particulate will be generated, that the most serious gaseous contaminants are Kr-85 and iodine.

The consequence of failing to meet this specification would be that discharges could be made to the environment which were not measured. This would not mean, ipso facto, that the limits of 4.1 and 4.2 had been exceeded but the precision of evaluation would be reduced and the values might have to be inferred from data taken at places other than at the discharge point.

5.2 PROCESS INSTRUMENTATION

APPLICABILITY

This specification establishes the operability requirements of process instrumentation.

OBJECTIVE

To assure that sufficient process instrumentation will be operable so that control of the process can be maintained.

SPECIFICATION

5.2.1 PRIOR TO OPERATION OF EITHER OF THE DISSOLVERS THE PRIMARY INSTRUMENTS SHOWN BELOW SHALL BE OPERABLE. IF ANY OF THE PRIMARY INSTRUMENTS FAIL DURING OPERATION, THE DISSOLVERS MAY BE OPERATED WITH THE LISTED ALTERNATES. IF ALL THE ALTERNATES FOR ANY PARTICULAR PRIMARY FAIL, IMMEDIATE ACTION WILL BE TAKEN TO SHUT THE DISSOLUTION OPERATION DOWN.

DISSOLVER 3C-1

PRIMARY INSTRUMENTATION	ACCEPTABLE ALTERNATE
3PRC-5	3PRC-5 (Manual)
	6PDR-3 (Off Gas Scrubber DP)
	Stack Monitor
LR-3	3DR-2
	3TRC-3
	3TRI-2

DISSOLVER 3C-2

PRIMARY INSTRUMENTATION	ACCEPTABLE ALTERNATE
3PRC-2	3PRC-2 (Manual)
	6PDR-3 (Off Gas Scrubber DP)
	Stack Monitor
LR-1	3DR-1
	3TRC-1
	3TRI-1

5.2.2 PRIOR TO OPERATION OF THE SOLVENT EXTRACTION SYSTEM THE PRIMARY INSTRUMENTS LISTED BELOW SHALL BE OPERABLE. IF ANY OF THE LISTED PRIMARY INSTRUMENTS FAIL DURING OPERATION OF THE SYSTEM, THE ALTERNATES ALSO LISTED BELOW MAY BE USED TO CONTINUE OPERATION. IF ANY OF THE ALTERNATE INSTRUMENTS FAIL WHEN BEING USED IN LIEU OF THE PRIMARY INSTRUMENT, THE NEXT ALTERNATE INSTRUMENT WILL BE USED.

IN THE EVENT ALL ALTERNATE INSTRUMENTS FAIL THE SOLVENT EXTRACTION SYSTEM WILL BE SHUT DOWN.

PRIMARY INSTRUMENT	ASSOCIATED STREAM OR EQUIPMENT	ACCEPTABLE ALTERNATE
14FRC-1	HAS	14D-1 HAS Make-up procedures.
4FRC-2	IIAX	4FRC-2 (Manual). Hand set air pressure on control valve. Spared pumps, 13G-11, 11A
4FRC-7	IAX	4FRC-7 (Manual). Hand set air pressure on control valve. Spared pumps 13G-11, 11A, 12, 12A
4FRC-12	IDX	4FRC-12 (Manual). Hand set air pressure on control valve. Spared pumps 13G-13, 11, 11A
4FRC-22	HAX	4FRC-22 (Manual). Hand set air pressure on control valve. Spared pumps 13G-11, 11A
4LRC-3	IIA Interface Control	4LRC-3 (Manual) 4LRC-3Y Hand set loading of pressure pot. 4PR-1
14FRC-2	HBX	14FRC-2 (Manual). Manual loading of control valve Depletion on tank 14D-6
14FR-11	IIBX	Positive displacement pump (14G-14) tank depletion rate. 14FAL-20

5.2.3 THE ION EXCHANGE COLUMNS WILL BE OPERATED AT CELL TEMPERATURE IF THE TEMPERATURE RECORDER CONTROLLER (TRC-4) IS INOPERABLE.

5.2.4 DURING OPERATION OF THE EVAPORATORS, THE PRIMARY INSTRUMENTATION SHOWN BELOW SHALL BE OPERABLE. IF ANY OF THE PRIMARY INSTRUMENTS FAIL DURING OPERATION, THE EVAPORATORS MAY BE OPERATED WITH THE ALTERNATE INSTRUMENTATION. IF THE ALTERNATES FOR A PARTICULAR PRIMARY FAILS, IMMEDIATE ACTION WILL BE TAKEN TO SHUT DOWN THE EVAPORATION SYSTEM.

LOW ENRICHED URANIUM PRODUCT EVAPORATOR

PRIMARY INSTRUMENT	ACCEPTABLE ALTERNATE
5DRC-20	5DRC-20 (Manual) 5DRC-23 on 5D-9 5DRC-23 (Manual)

REWORK EVAPORATOR

PRIMARY INSTRUMENT	ACCEPTABLE ALTERNATE
7LRC-4	7LRC-4 (Manual) 7DR-4 7TRI-5 Volume and analysis of batch from 7D-8

5.2.5 PRIOR TO OPERATION OF THE FEED ADJUSTMENT AND ACCOUNTABILITY TANK, THE PRIMARY INSTRUMENTS SHOWN BELOW SHALL BE OPERABLE. IF ANY OF THE PRIMARY INSTRUMENTS FAIL DURING OPERATION, THE TANK MAY BE OPERATED WITH THE LISTED ALTERNATES. IF ANY OF THE ALTERNATES FAIL, IMMEDIATE ACTION WILL BE TAKEN TO SHUT DOWN THE FEED ADJUSTMENT AND ACCOUNTABILITY SYSTEM.

PRIMARY INSTRUMENT	ACCEPTABLE ALTERNATE
DENSITY RECORDER DR-4	DENSITY INDICATOR DI-3
LEVEL RECORDER LR-6	LEVEL CONTROL LOW LCL-8 3TI2-5

BASES

The operations in this plant, as in any chemical plant, are controlled by a variety of process instruments. Because of the need to maintain close control of many process variables, a multiplicity of instruments have been included in the design of this plant. In some cases important instruments are provided in duplicate. In others, alternate ways of determining the desired information have been provided. The entire process instrumentation has been reviewed and this specification enumerates, for the process steps important to public health and safety, the primary and secondary control instrumentation, and spells out the instrumentation which must be operable at the start of operations and that which must be maintained in order to continue operations.

Selection of primary and secondary instruments on the dissolvers 3C-1 and 3C-2 is based upon two safety considerations: (1) The dissolver off gas being routed through the proper scrubbing equipment (i. e., the DOG system), and (2) the solution in the dissolver not being overconcentrated with a resultant possibility of a criticality incident. The primary instrument on the dissolvers for metering off gas routing is 3 PRC-5 on 3C-1 and 3PRC-2 on 3C-2. These instruments are recording, controlling instruments operating on the pressure in the dissolvers. Should the automatic feature on the instruments fail, both instruments can be switched to manual control. With this setting more operator attention is required but control is still possible.

Should the indicating part of these instruments fail then normal control is no longer possible. In this instance the off-gas scrubber differential pressure instrument 6PDR-3 can be used to indicate if the proper gas flow is leaving the dissolvers and consequently the proper pressure is being maintained in the dissolvers. Should 6PDR-3 fail the stack monitor will indicate whether the DOG is being routed through the DOG system. If the dissolvers pressurize, DOG will leak into the CPC ventilation air which will be picked up by the stack monitor. A high reading thus indicates a high DOG pressure and the pressure control valve on the dissolvers should be opened. Overconcentration of dissolver solution will be detected on the primary instrument LR-3 or LR-1. Should this instrument fail 3DR-2 or 1 will indicate excessive concentration and should LR and DR instruments both fail 3TRC-1 and 3 will indicate overconcentration. A ten per cent increase in uranium concentration in a dissolver product of 1 M U and 2 M HNO_3 represents approximately 1.8°C change in boiling point.

Instrument requirements for the solvent extraction columns are set to minimize any possibility of fissionable material concentration build-up to unsafe levels by reflux or abnormal operation as a consequence of instrument failure. Conditions which could possibly lead to high concentrations are : omission of HNO_3 from the HAS scrub stream, stoppage of the IIAX stream (recycle from the Pu ion exchange columns can introduce high Pu concentrations into this column), failure of ferrous sulfamate addition to the HBX column, and stoppage of IIBX flow rate to the IIB column. Variables requiring control to prevent recycle with possible subsequent problems, but not necessarily criticality problems are also given. These variables are the IAX, IDX and HAX flows and the IIA interface. On such streams it is possible to operate by throttling a pump discharge valve, in these instances the pumps themselves are listed as acceptable alternates.

A high degree of control on the high enriched evaporator is not necessary due to the safe (by poison) nature of evaporator itself and all the vessels following this evaporator in the system.

The Low Enriched Evaporator does have non-safe tanks downstream from the product receiver. Consequently, the primary and alternative control instruments are listed on this evaporator. The Rework Evaporator is safe only by concentration control so the requirement for minimum instrumentation here is obvious.

The Feed and Accountability tank is safe only by concentration control. Instrumentation is listed here to prevent over concentrations.

The consequence of losing any single instrument is to reduce the margin of safety in operations, or more properly the defense in depth. This specification represents a definition of the minimum acceptable defense in depth and if it were violated it would represent the operation of the plant in a "blind" condition. Because of the character of the system and its inertia, this would not necessarily result in any incident but this would represent an unacceptable margin of safety.

5.3 VENTILATION

APPLICABILITY

This specification establishes the operability requirements of the ventilation systems.

OBJECTIVE

To assure that proper pressure differentials will be maintained between operating equipment and equipment environment and between adjacent processing areas which will assure, to a high degree of probability, the proper processing of all gaseous effluents within the facility.

SPECIFICATION

5.3.1 DISSOLVER OFF-GAS SYSTEM (DOG)

FOR THE FIRST HOUR OF ANY BATCH DISSOLUTION OR DURING SHEAR OPERATIONS IN THE PMC, THE MINIMUM AMOUNT OF OPERABLE EQUIPMENT SHALL INCLUDE A CHEMICAL SCRUBBER AND ONE DISSOLVER OFF-GAS FILTER AND ITS ASSOCIATED BLOWER UNLESS THE FUEL BATCH CONTAINS IN EXCESS OF 0.16 CURIE OF IODINE-131 IN WHICH CASE THE OPERABLE EQUIPMENT SHALL INCLUDE IN ADDITION ONE DISSOLVER OFF-GAS HEATER AND ONE DISSOLVER OFF-GAS SILVER NITRATE REACTOR. IF THIS MINIMUM REQUIREMENT IS NOT MET, THE DISSOLVER SHALL BE IMMEDIATELY COOLED AND SHEARING OPERATIONS SHALL BE IMMEDIATELY DISCONTINUED.

5.3.2 VESSEL OFF-GAS SYSTEM (VOG)

THE VESSEL OFF-GAS SYSTEM SHALL BE OPERABLE AT THE START OF CAMPAIGN OPERATIONS INVOLVING THE FEED ADJUSTMENT AND ACCOUNTABILITY TANK. THE EQUIPMENT SHALL BE OPERABLE THROUGHOUT A CAMPAIGN EXCEPT UNDER CONDITIONS DEFINED BELOW.

5.3.2.1 OPERATIONS MAY CONTINUE FOR 60 DAYS WITH ONLY ONE OFF-GAS FILTER AND ITS ASSOCIATED BLOWER OPERABLE.

5.3.2.2 OPERATIONS MAY CONTINUE FOR 24 HOURS IF BOTH FILTERS AND/OR THEIR ASSOCIATED BLOWERS FAIL.

5.3.3 MAIN VENTILATION SYSTEM

THE ENTIRE MAIN VENTILATION EXHAUST SYSTEM SHALL BE OPERABLE AT THE START OF A CAMPAIGN. THE EQUIPMENT WILL BE OPERABLE THROUGHOUT THE CAMPAIGN EXCEPT UNDER CONDITIONS DEFINED BELOW.

5.3.3.1 OPERATIONS MAY CONTINUE FOR 60 DAYS WITH ONLY ONE MAIN EXHAUST FILTER AND ITS ASSOCIATED BLOWER OPERABLE.

5.3.3.2 IF BOTH MAIN EXHAUST FILTER SYSTEMS BECOME INOPERATIVE, IMMEDIATE AND CONTINUING ATTENTION SHALL BE DIRECTED TOWARD ORDERLY PLANT SHUT DOWN.

5.3.3.3 THE INTAKE AIR SUPPLY SHALL BE SHUT OFF IF THE EXHAUST SYSTEM IS SHUT DOWN.

5.3.4 WASTE TANK OFF-GAS SYSTEM

IF A FILTER, BLOWER OR A WASTE TANK OFF-GAS CONDENSER FAILS, OPERATIONS SHALL BE SWITCHED TO THE APPROPRIATE STAND-BY EQUIPMENT AND IMMEDIATE REPAIR SHALL BE INSTITUTED TO EXPEDITE THE RETURN OF THE FAILED EQUIPMENT TO SERVICE.

BASES

The primary control of airborne radioactivity in the plant is provided by four ventilation systems. These systems provide proper pressure differentials between operating equipment and occupied areas so that plant personnel are protected from airborne activity. They also provide treatment systems for the removal of radioactivity from the ventilation streams prior to their release to the atmosphere. Because of the importance of these systems this specification has been designed to assure the availability at all times of sufficient ventilation capacity to provide the degree of protection specified in the design of this plant.

The DOG system treats gaseous effluents from the dissolver, the bundle shear and the pin shear. The off-gas from the shearing operations is connected to the DOG system upstream from the chemical scrubber and the combined dissolver and shear off-gases pass through the scrubber and then can be either routed through a reheater, a silver nitrate reactor, a cooler, and a filter prior to blower discharge into the stack, or the gas can be routed directly from the chemical scrubber to the filter. Actual routing of the gas will depend on the Iodine-131 content of the fuel batches.

The chemical scrubber removes 90 percent of the Iodine-131 and if necessary, the chemical scrubber in combination with the silver nitrate reactor removes 99.5 percent of the Iodine-131. To assure the safe discharge of gaseous effluents during the initial period of each batch dissolution, if the minimum operable equipment requirements are not met, measures will be taken to reduce the evolution of radioiodine by discontinuing the shearing operation and by cooling the dissolver, thereby slowing or stopping the reaction.

The VOG system provides equipment for the treatment of the gaseous effluents from radioactivity-containing process vessels throughout the plant. This gaseous effluent is not considered to have radioiodine, consequently no silver nitrate reactors are provided. The gases are condensed, scrubbed, heated and filtered prior to final blower discharge to the stack. If both VOG filter-blower units fail, operations may continue for a period since negligible radioactivity is normally released from the system and measures can be taken such as the reduction of air sparging rates to reduce the volume expelled through the system. An arbitrary limit of 24 hours of such operation has been selected as sufficient to permit repair.

The Main Ventilation System provides for the filtration of intake air, the controlled passage of air through the process building from inactive areas to successively more active areas, and the filtration of discharge air prior to exhaust up the stack. In order to provide for continuous operation, two parallel filter-exhauster units are provided. Only one of these exhaust systems is necessary to provide sufficient ventilation for facility. Accordingly, if either one of the exhaust units failed, operations could be allowed for an indefinite period. To insure continuing maintenance attention, an arbitrary operating time limit of 60 days has been imposed with only one exhaust system operable.

If both exhaust systems fail, the influent air supply will be manually shut off in order to minimize the possibility of expelling contaminants from the process building.

The waste tank off-gas system condenses and filters gaseous effluents from the storage tanks. In this system, dual equipment is provided (i.e. condensers, filters and blowers). Failure of an in-line component will require a switch to the alternate standby component. Since a failure would decrease the defense in depth, high priority shall be given to the repair of a failed component. A set time in

which to complete repairs to an initial in-line component is somewhat meaningless. Assuming the presence of waste in a tank and assuming further the failure of one filter-blower set or even both presents situation wherein speed must be exercised to complete repairs; the waste cannot be (reasonably) removed from the tank.

The consequence of violating this specification would be to risk the exposure of operating personnel and the general public to levels of airborne contamination in excess of the normally expected acceptable levels as defined in paragraphs 7.6 through 7.9 of the Safety Analysis. In paragraphs 7.30 - 7.36 the consequences of various possible incidents which would result from failure to follow this specification have been evaluated to show that these events are acceptable within the guidelines of 10 CFR Part 100.

6.0 SURVEILLANCE REQUIREMENTS

The specifications listed in this section require inspections of certain equipment or systems which, with one exception, are not primary safeguards but which are desirable for a defense in depth if a primary system or safeguard fails. The lone exception is the specification (6.1.1) requiring surveillance of the boron glass Raschig rings which are a primary safeguard in vessels 5D-13A, B and C and for which due regard is given. The other specifications in this section are primarily intended to focus attention on aspects of plant operations which, while not primary safeguards, are of sufficient importance to direct, generally, immediate and continuing action to the return to service of the failed component. If inspection required by specifications 6.2 through 6.5 reveals the inoperability of any of the specified equipment plant operation may continue under "ready conditions".

6.1 BORON RASCHIG RINGS

APPLICABILITY

This specification establishes requirements for determining periodically the content and/or distribution of boron in Raschig rings.

OBJECTIVE

To assure an adequate distribution of boron within process vessels which may contain concentrations of fissionable isotopes which in the absence of nuclear poison would be unsafe.

SPECIFICATION

6.1.1 VESSELS 5D-13 A, B AND C SHALL BE INSPECTED ANNUALLY TO DETERMINE THAT THE PACKED HEIGHT OF BORON GLASS RASCHIG RINGS HAS BEEN MAINTAINED. IF THE PACKED HEIGHT HAS DECREASED ONE RING THICKNESS, REPLACEMENT RINGS SHALL BE ADDED TO COMPLETELY FILL THE VESSEL.

6.1.2 WHEN THE VESSELS LISTED IN 6.1.1 ABOVE ARE INSPECTED, A REPRESENTATIVE SAMPLE OF THE RASCHIG RINGS SHALL BE TAKEN TO DETERMINE THAT THERE IS A CONCENTRATION OF 20 GRAMS OF BORON PER LITER OF VESSEL PACKED VOLUME. IF THE CONCENTRATION IS BELOW THIS VALUE, RINGS SHALL BE REPLACED.

6.1.3 CALIBRATION OF OTHER VESSELS CONTAINING BORON GLASS OR BORON STAINLESS STEEL RASCHIG RINGS SHALL BE PERFORMED EVERY TWO YEARS TO DETERMINE THAT THE VOLUME OF THE RINGS PER UNIT OF PACKED HEIGHT IS MAINTAINED WITHIN 20 PERCENT OF THE DESIGN VALUE AND TO DETERMINE THAT THE RINGS ARE ABOVE THE OVERFLOW. IF THESE CONDITIONS ARE NOT MET, RING REPLACEMENT SHALL BE MADE UNTIL THE CONDITIONS ARE SATISFIED.

BASES

In the high enriched uranium product vessels (5D-13A, B, and C) boron glass Raschig rings are used as a primary safeguard to maintain subcriticality; hence, a chemical determination of the boron content will be made on an arbitrarily set yearly basis. In the course of sampling the Raschig rings, a physical inspection will also be made which may reveal a need for ring replacement. These vessels could hold enriched uranium at concentrations in excess of 400 g/l. In the case of other process vessels containing boron glass or boron stainless steel Raschig rings, (letter from NFS to USAEC-DML dated 3/3[illegible]/65) the presence of the rings is a secondary

deterrent against the possibility of a critical incident. In these vessels, control of the concentration of fissile material serves as the primary safeguard. The boron content of the glass or stainless steel Raschig rings in these tanks greatly exceeds requirements, accordingly, it is not deemed necessary to perform a chemical evaluation of the boron content of these rings.

It is unlikely that a criticality incident could occur in the High Enriched Evaporator (5C-5). The vapor space in the top of the evaporator will be filled with glass Raschig rings to preclude the possibility of a nuclear incident if concentrated fissile material were to enter this enlarged space. The design of the evaporator is such that even in the event of the failure of the Raschig ring support plate, the volume of rings could not drop below the height of the overflow line.

The consequence of failing to meet the requirements of this specification is to reduce the level of confidence in the presence and efficacy of the neutron poisons provided. The specification is designed to assure that on a regular basis the indicated safety inspections will be carried out and necessary remedial action taken. Normally it will be expected that inspection will show that no remedial work is required.

6.2 SUMP ALARMS AND EDUCTORS

APPLICABILITY

This specification establishes the surveillance requirements for sump eductors and level alarms.

OBJECTIVE

To assure that liquid accumulation will be detected and can be removed.

SPECIFICATION

6.2.1 THE SUMP ALARM AND TRANSFER EDUCTOR IN THE PPC, XC-2 AND XC-3 SHALL BE CHECKED AS TO OPERABILITY ONCE A MONTH OR BETWEEN CAMPAIGNS, WHICHEVER IS LONGER. IF ANY SUMP EDUCTOR OR LEVEL ALARM IS INOPERATIVE, IT MUST BE REPAIRED PRIOR TO THE START OF THE NEXT CAMPAIGN OPERATIONS IN THE AREA IN WHICH SUCH DEFECTIVE COMPONENT(S) IS LOCATED.

BASES

A discussion of sump operational philosophy has been given in a letter from NFS to the US Atomic Energy Commission, Division of Materials Licensing, dated February 17, 1965, p. 7. A listing of sumps may be found as Attachment B to a letter from NFS to US Atomic Energy Commission, Division of Materials Licensing, dated May 3, 1965.

Since the geometry of the floors and sumps are such that they can accommodate the total contents of the largest vessel in a particular area, the operability of the sump level alarms and detectors serves only as a secondary defense against criticality; that is, it would take the catastrophic bottom rupture of two tanks, together containing fissile material in sufficient quantity to create a nuclear incident to precipitate same. It is expected the loss of a vessel's contents would be detected from collected process data. Accordingly, the frequency of inspecting the sump alarm and eductor equipment need be no more stringent than indicated in the specification. Further, only in PPC, XC-2 and XC-3 is there a concentration of fissionable material in any tank high enough to cause a critical incident even if the sump were filled.

The consequence of failing to meet the requirements of this specification is to reduce the level of confidence in the operability of the sump eductor system. The specification is designed to assure that on a regular basis the indicated safety inspections will be carried out.

6.3 INSTRUMENTS IN ANNULAR SPACES SURROUNDING WASTE STORAGE TANKS

APPLICABILITY

This specification establishes the surveillance requirements for the instruments within the annular space surrounding the liquid waste storage tanks.

OBJECTIVE

This surveillance requirement sets the frequency, which will assure, to a high degree of probability, that liquid in the annular space is detected.

SPECIFICATION

6.3.1 THE LIQUID LEVEL MONITORING SYSTEM FOR THE ANNULAR SPACE SURROUNDING THE WORKING LIQUID WASTE STORAGE TANKS SHALL BE CHECKED FOR OPERABILITY AT LEAST ONCE PER MONTH AND, IF FOUND INOPERABLE, IMMEDIATE AND CONTINUOUS EFFORT SHALL BE DIRECTED TO EFFECT ITS REPAIR.

BASES

In the design of the waste storage system, barriers for the protection of the public have been established in the form of the mild steel primary containment vessel, a mild steel catch pan, a poured concrete vault, and the barrier provided by the geology of the burial site itself. The space between the primary containment vessel and the catch pan is the annular space to which reference is made.

The annular space is provided with a level indicator, a high level alarm, and a discharge pump. Wastes are neutralized to $p^H>7$ before storage to minimize corrosion which might result in tank failure. Recognizing, however, that tank leakage could occur, the annular space provides for hold-up and sampling. The purpose of the high level alarm in the annular space is to annunciate any major leakage.

Inasmuch as leakage from the vault, should the pan overflow, would require a considerable time period--on the order of years-- to migrate away from the immediate vicinity of the tank, a monthly surveillance frequency of the detection alarms provides an adequate margin of safety.

The consequence of failing to meet the requirements of the specification is to reduce the level of confidence in the operability of the liquid monitoring system in the annular space around the waste tanks. This specification is designed to assure that on a regular basis the indicated safety inspections will be carried out.

6.4 EMERGENCY UTILITY EQUIPMENT

APPLICABILITY

This specification establishes the surveillance requirements for the stand-by utility equipment.

OBJECTIVE

To assure that stand-by utility equipment deemed necessary for continued facility operations is maintained in an operable condition.

SPECIFICATION

6.4.1 THE EMERGENCY GENERATOR (30T-1), AIR COMPRESSOR (31K-1A) AND THE COOLING WATER PUMP (32G-4B) SHALL BE OPERATED AT LEAST ONCE EVERY THREE(3) MONTHS TO DETERMINE: 1. THE AUTOMATIC START CAPABILITY AND 2. PERFORMANCE FOR AT LEAST ONE HOUR UNDER AN OPERATING LOAD. IF ANY OF THE ABOVE EQUIPMENT IS FOUND TO BE DEFICIENT, IMMEDIATE AND CONTINUING ACTION SHALL BE TAKEN TO RECTIFY THE CONDITION. IF NOT RETURNED TO SERVICE IN 90 DAYS, THE NEXT FOLLOWING CAMPAIGN SHALL NOT START.

6.4.2 ANY OF THE EQUIPMENT LISTED BELOW WHICH IS NOT USED IN A 3-MONTH PERIOD SHALL BE TESTED AS ABOVE, EXCEPT FOR AUTOMATIC START PROVISION, WITH SIMILAR PROVISIONS PROVIDED FOR ANY DEFICIENCIES FOUND.

BOILER FEED PUMPS 31G-2A AND 31G-2B
BOILER DRAFT FANS 31K-2 and 31K-2A
PLANT WATER PUMPS 32G-2A AND 32 G-2B

6.4.3 THE QUANTITY OF DIESEL FUEL IN THE OIL STORAGE TANK (31D-2) SHALL, DURING NORMAL OPERATIONS, BE MAINTAINED AT 8000 GALLONS OR MORE OF DIESEL FUEL. IF USED DURING ABNORMAL CONDITIONS, OPERATIONS MAY CONTINUE AS LONG AS A VOLUME OF 4000 GALLONS OR GREATER IS MAINTAINED. A VOLUME LESS THAT THIS LATTER VALUE WILL REQUIRE THE COMMENCEMENT OF PLANT SHUT DOWN WHICH SHALL CONTINUE UNLESS THE PRIMARY FUEL (NATURAL GAS) SERVICE IS RESTORED.

BASES

This plant, like any industrial complex, requires a number of utility services, such as electricity, gas, steam, air and water, in order to run. Most of the many various safety equipment and devices installed in the plant require one or more of these services for continued operation. Therefore, stand-by equipment has been provided for use in the event of the failure of a utility-provided

service. This equipment, in order to ensure satisfactory performance of its intended function, shall be tested on a routine basis. In the event of natural gas outage, diesel fuel has been provided in sufficient capacity to assure normal operations for about 10 hours. Unless weather conditions are very severe, diesel fuel can be replenished within 10 hours thereby permitting continued operations.

The consequence of failing to meet the requirements of this specification is to reduce the level of confidence in the operability of the stand-by services. This specification is designed to assure that on a regular basis the stand-by systems are tested and their operability proved.

6.5 AIR FILTRATION EQUIPMENT

APPLICABILITY

This specification establishes surveillance requirements for the final air filters in the off-gas and ventilation systems.

OBJECTIVE

To assure proper performance of the final air filters used in the plant.

SPECIFICATION

6.5.1 IF ANY OF THE FILTERS SHOWN BELOW ARE CHANGED OR MODIFIED, THEY SHALL BE TESTED, PRIOR TO SERVICE, WITH PARTICULATES OF AN APPROPRIATE SIZE TO ESTABLISH THAT THE INSTALLED FILTERS PROVIDE A COLLECTION EFFICIENCY EXCEEDING 99.95% FOR PARTICULATES 0.3 MICRONS IN DIAMETER OR LARGER.

SYSTEM	FILTERS
DISSOLVER OFF-GAS	6T-1, 6T-1A
VESSEL OFF-GAS	6T-2, 6T-2A
WASTE TANK OFF-GAS	8T-1, 8T-1A
BUILDING VENTILATION	15T-49, 15T-49A

BASES

The final filters in the ventilation and off-gas systems are the primary protection against the spread of particulate contamination during normal operations and particularly under accident conditions. It has been observed at AEC sites that testing is necessary for assurance of filter performance after installation to detect by-pass leakage as well as filter damage during storage and handling. Repeated tests are not necessary as there are manometers and radioactivity monitors which will indicate filter integrity and performance after the filter is placed on stream.

These tests are conducted by introducing a known amount of an appropriate test material upstream of the filter being tested and taking samples downstream of the filter to determine the removal efficiency.

The consequence of failing to meet the requirements of this specification is to risk the introduction of a faulty filter. In any case, the stack monitor would quickly show up the existence of the fault and force its correction.

6.6 DISSOLVER DILUTION AIR

APPLICABILITY

This specification establishes the surveillance requirements of the air sparging systems for the dissolvers.

OBJECTIVE

To assure that the concentration of hydrogen gas in the dissolver off-gas is below its lower explosive limit.

SPECIFICATION

6.6.1 PRIOR TO THE DISSOLUTION OF ZIRCONIUM ALLOY FUELS THE AIR SPARGING SYSTEM SHALL BE TESTED TO ASSURE THAT SUFFICIENT AIR SHALL BE ADDED DURING DISSOLUTION TO MAINTAIN THE HYDROGEN GAS CONCENTRATION BELOW ITS LOWER EXPLOSIVE LIMIT.

BASES

The presence of dilution sparge air constitutes, together with the dissolvent acid addition rate, a method of insuring against the possibility of a hydrogen explosion.

During the dissolution of zirconium alloy fuels, there could be sufficient hydrogen gas generated from the chemical reaction of the metal and the acid to produce a hydrogen gas concentration downstream of the dissolver's reflux condenser greater that the lower explosive limit of hydrogen. Accordingly, the dissolver must be purged with dilution air to maintain the concentration of hydrogen gas below its explosive limit.

The consequence of failing to make the check required by this specification is to reduce the level of confidence that sufficient air is available to assure that an explosive mixture of hydrogen and air cannot form in the dissolver during dissolution of uranium-zirconium alloy. Explosive mixture, if formed, does not have a ready mechanism for ignition. Further, the concentration would be reduced well below the lower explosive limit as soon as the DOG stream is mixed with the general ventilation air in the stack.

6.7 BORIC ACID

APPLICABILITY

This specification establishes the surveillance requirements for the boric acid in tank 14D-32.

OBJECTIVE

To assure that boric acid will be available in the event of criticality incident in either the dissolver or the rework evaporator.

SPECIFICATION

6.7.1 TANK 14D-32 SHALL BE INSPECTED PRIOR TO EACH CAMPAIGN TO DETERMINE THE PRESENCE OF BORIC ACID (AT LEAST 22 g/l H_3BO_3) SOLUTION AND THE SOLUTION SHALL BE AVAILABLE FOR USE THROUGHOUT THE CAMPAIGN.

BASES

The "drown tank", 14D-32, has been provided to quench or prevent criticality excursions in the dissolver or the re-work evaporator by the rapid addition of boric acid to those vessels. If an excursion occurs, the "drown tank" is the principal aid in minimizing the extent of the excursion. The data of ORNL-3309 Soluble Neutron Poisons as a Primary Criticality Control in Shielded and Contained Radiochemical Facilities Table II, Page 21 for nitric acid-uranyl nitrate solutions expected in the dissolver indicate that the boric acid concentration will be such that a precipitate will not be formed in the dissolver.

The consequence of failing to meet the requirements of this specification is to reduce the level of confidence that the backup presence of quench boric acid solution is indeed available. This specification is designed to assure that on a regular basis the indicated safety inspection is made.

6.8 BLANKING-OFF AND LOCKING-OUT

APPLICABILITY

This specification establishes the surveillance requirement of equipment which must be blanked-off or locked-out during processing operations.

OBJECTIVE

To assure that flows are properly routed during a campaign thereby avoiding accidental criticality through precipitation or over concentration of fissionable material.

SPECIFICATION

6.8.1 CONFORMANCE WITH THE FOLLOWING SPECIFICATIONS SHALL BE DETERMINED BY INSPECTION OF A SHIFT SUPERVISOR OR HIGHER SUPERVISION.

6.8.1.1 IMMEDIATELY PRIOR TO EACH HIGH ENRICHED URANIUM CAMPAIGN, CONNECTIONS TO VESSELS INTENDED FOR USE ONLY DURING LOW ENRICHED URANIUM CAMPAIGNS SHALL BE BLANKED OFF.

6.8.1.2 DECONTAMINATION SYSTEMS SHALL BE BLANKED OFF BEFORE ANY OPERATION OF THE PROCESS SYSTEM SERVED BY THAT DECONTAMINATION SYSTEM.

6.8.1.3 ALL PIPING LINES WHICH CAN INTRODUCE PRECIPITANTS INTO VESSELS CONTAINING SOLUTIONS OF FISSIONABLE ISOTOPES SHALL BE BLANKED-OFF OR LOCKED-OUT AFTER EACH USE.

BASES

Specification 6.8.1.1 has been established to prevent the inadvertent concentration of high-enriched uranium in the low enriched uranium evaporator because concentrated high-enriched uranium would not be nuclearly safe in the geometrically unfavorable silica gel columns which are downstream from the low enriched uranium evaporator. Plutonium and uranium can be precipitated from process solutions by the addition of caustic, sodium carbonate, or other precipitants. Inadvertent addition of caustic could result in criticality. Accordingly, the specifications requiring blanking-off or locking-out of systems through which such an inadvertent caustic addition could be made have been established.

The use of lock-out procedures and devices is discussed in the letter, Nuclear Fuel Services, Inc. to the United States Atomic Energy Commission, Division of Materials Licensing, dated March 31, 1965.

The consequence of failing to meet the requirements of this specification 6.8.1.1 is to risk the introduction of highly enriched uranium solution into equipment which is designed for handling low enriched fuel. The possibility of a critical incident is thus significantly increased. Similarly specifications 6.8.1.2 and 6.8.1.3 militate against the precipitation of uranium which in turn would increase the chance of a critical incident. Therefore, strong administrative controls are set up and monitored by this specification to assure that the necessary blanks and lock-outs are installed prior to the operations indicated.

6.9 STEAM PRESSURE

APPLICABILITY

This specification establishes surveillance requirements for steam pressure used for evaporation.

OBJECTIVE

To assure that the temperature for autoignition of entrained organic degradation products will not be reached in the evaporators.

SPECIFICATION

6.9.1 PRIOR TO THE EVAPORATION OF ANY PROCESS SOLUTION, THE PRESSURE OF THE STEAM USED FOR EVAPORATION IN ALL EVAPORATORS EXCEPT THE GENERAL PURPOSE EVAPORATOR SHALL BE CHECKED TO DETERMINE THAT IT DOES NOT EXCEED 25 PSIG.

BASES

"Red oil", an organic phase mixture of uranyl nitrate, tributyl phosphate, dibutyl phosphate, and other organic decomposition products can be formed under certain conditions if organic products are carried over into the evaporator and allowed to concentrate. Red oil has been found to be temperature sensitive, and can explode at temperatures exceeding approximately 274°F. While this material can be formed only under a series of unusual maloperations of the process upstream, it is simple to provide an absolute protection from the possibility of explosion by limiting the pressure of the steam supplied to the evaporators thus maintaining the temperature below 267°F.

The General Purpose Evaporator is excluded from this specification because no particular hazard results from its operation at higher steam pressures, since it is not possible for "red oil" to be formed from any of the streams that would be processed in the evaporator.

The consequence of failing to meet the requirement of this specification is to reduce, or possibly remove, the margin of safety provided to prevent a possible "red oil: explosion by limiting the temperature to below that required to initiate the reaction. Such an explosion would not necessarily occur if the 274F temperature were exceeded but it now becomes a possibility...which should be avoided.

7.0 ADMINISTRATIVE REQUIREMENTS

In this section there are specified requirements which are essential to the maintenance of safety in operations, but which by their nature are not readily expressed in quantitative terms. These are identified as administrative specifications, governing both administrative procedures and certain operating procedures.

7.1 ADMINISTRATIVE REQUIREMENTS

APPLICABILITY

This specification establishes administrative standards for governing the operation of the facility.

OBJECTIVE

To assure that a management system responsive to the safety needs of the operation is established and maintained.

SPECIFICATION

7.1.1 ORGANIZATION. FOR PURPOSES OF MAINTAINING SAFE OPERATION AND CONTROL OF THE FACILITY AND OF ATTENDANT ACTIVITIES, NUCLEAR FUEL SERVICES, INC. SHALL PROVIDE MANAGEMENT OF THE PLANT THROUGH AN ADMINISTRATIVE FRAMEWORK INCLUDED, BUT NOT LIMITED TO, THE FOLLOWING:

7.1.1.1 A GENERAL MANAGER, RESPONSIBLE FOR ALL ACTIVITIES AT THE PLANT INCLUDING HEALTH AND SAFETY, NUCLEAR SAFETY, AND ADHERENCE TO THE LIMITS AND CONDITIONS SET FORTH IN THIS LICENSE.

7.1.1.2 A PRODUCTION MANAGER, REPORTING TO THE GENERAL MANAGER, AND RESPONSIBLE FOR CARRYING OUT PRODUCTION ACTIVITIES IN ACCORDANCE WITH APPROVED PROCEDURES AND ACCEPTED HEALTH AND SAFETY STANDARDS.

7.1.1.3 A HEALTH AND SAFETY DIRECTOR, REPORTING TO THE GENERAL MANAGER, AND RESPONSIBLE FOR PLANT SAFETY AND FOR REVIEW OF ALL PLANT ACTIVITIES TO ASSURE CONFORMANCE WITH THE HEALTH AND SAFETY MANUAL AND THE STANDARD OPERATING PROCEDURES.

7.1.1.4 A TECHNICAL SERVICES MANAGER, REPORTING TO THE GENERAL MANAGER, AND RESPONSIBLE FOR THE TECHNICAL REVIEW OF ALL OPERATIONS TO ASSURE THE TECHNICAL SOUNDNESS OF PROPOSED OPERATIONS AND IN PARTICULAR TO ASSURE THE MAINTENANCE OF A NUCLEARLY SUBCRITICAL SYSTEM.

7.1.1.5 A PLANT SAFETY COMMITTEE, RESPONSIBLE FOR REVIEW AND APPROVAL OF ALL STANDARD OPERATING PROCEDURES, CHANGES IN THE PROCESS, THE PROCESS SYSTEM, AND STANDARD OPERATING PROCEDURES, OPERATIONS EXCEEDING TECHNICAL SPECIFICATION LIMITS, AND LETTERS OF AUTHORIZATION TO ASSURE SAFE OPERATIONS; THE COMMITTEE TO CONSIST OF, BUT NOT TO BE LIMITED TO, THE GENERAL MANAGER, TECHNICAL SERVICES MANAGER, HEALTH AND SAFETY DIRECTOR, AND THE PRODUCTION MANAGER.

7.1.2 PROCEDURES SHALL BE MAINTAINED UNDER THE DIRECTION OF THE GENERAL MANAGER GOVERNING THE:

7.1.2.1 PREPARATION, APPROVAL, ISSUANCE OF ALL OPERATING INSTRUCTIONS AND CHANGES THERETO, INCLUDING, BUT NOT LIMITED TO, STANDARD OPERATING PROCEDURES, LETTERS OF AUTHORIZATION, RUN SHEETS, SPECIAL WORK PROCEDURES, AND EXTENDED WORK PROCEDURES WHERE NUCLEAR CRITICALITY OR RADIATION SAFETY ARE CONSIDERATIONS.

7.1.2.2 INVESTIGATION OF ABNORMAL CONDITIONS WHICH INVOLVE THE PROCESSING, HANDLING OR STORAGE OF RADIOACTIVE MATERIALS AND WHICH AFFECT NUCLEAR CRITICALITY OR RADIATION SAFETY.

7.1.2.3 ACTIONS TO BE TAKEN IN EVENT OF AN EMERGENCY INVOLVING RADIOACTIVE MATERIAL;SUCH ACTIONS TO INCLUDE, BUT NOT LIMITED TO, SHUTDOWN OF SPECIFIC EQUIPMENT, ACTIVATION OF THE EMERGENCY PLAN, SUMMONING OF OUTSIDE SUPPORT, AND REENTRY TO THE ACCIDENT AREA.

7.1.2.4 PERFORMANCE OF PERIODIC REVIEWS OF OPERATING PRACTICES, RECORDS AND AUDITS.

7.1.3 PLANT OPERATION. NUCLEAR FUEL SERVICES' MANAGEMENT SHALL BE RESPONSIBLE FOR PLANT OPERATION. A COMPLETE AND CURRENT SET OF OPERATIONAL PROCEDURES SHALL BE PROVIDED IN THE CONTROL ROOM AND IN OTHER APPROPRIATE AREAS WHERE CONTROL FUNCTIONS ARE PERFORMED.

7.1.4 DETAILED WRITTEN PROCEDURES, RUN SHEETS, LETTERS OF AUTHORIZATION, AND EXTENDED WORK PROCEDURES SHALL BE PROVIDED FOR OPERATION OF THE PLANT. IN ADDITION, PROCEDURES FOR EMERGENCY OPERATIONS AND FOR RADIATION PROTECTION SHALL BE PROVIDED.

7.1.5 LOGS AND RECORDS. LOGS AND RECORDS SHALL BE MAINTAINED AS REQUIRED BY THE PROVISIONAL LICENSE AND SHALL ALSO INCLUDE DETAILS OF ANY ABNORMAL OCCURRENCES AS DEFINED BY PLANT MANAGEMENT AFFECTING THE SAFETY OF THE PLANT AND ACTION TAKEN THEREON.

7.1.6 IF ANY TECHNICAL SPECIFICATION LIMIT IS EXCEEDED, THE FOLLOWING PROCEDURES WILL BE OBSERVED:

7.1.6.1 IMMEDIATE STEPS SHALL BE TAKEN IN THE OPERATION OF THE SYSTEM INVOLVED TO REMOVE THE CAUSE OF EXCEEDING THE SPECIFICATION OR TO RELIEVE ITS EFFECT, AS APPROPRIATE. OPERATION MAY CONTINUE IN A "READY CONDITION" TEMPORARILY UNLESS THERE IS ANOTHER OCCURRENCE OR INDICATION OF AN OCCURRENCE THAT WOULD FURTHER REDUCE THE MARGIN OF SAFETY IN WHICH EVENT IMMEDIATE SHUTDOWN IS REQUIRED.

7.1.6.2 IMMEDIATE STEPS WILL BE TAKEN TO ESTABLISH THE CAUSE OF THE OCCURRENCE IN 7.1.6.1 BY RESPONSIBLE MANAGEMENT REVIEW.

7.1.6.3 NOTWITHSTANDING ANY OTHER PROVISION OF THIS LICENSE THE PLANT SHALL BE SHUTDOWN AND NORMAL OPERATION MAY NOT BE RESUMED PRIOR TO AEC REGULATORY CONCURRENCE WHEN SPECIFICATION 4.1.1 OR 4.2.1 HAS BEEN EXCEEDED.

7.2 PROCEDURES FOR UTILIZING RESPIRATORY PROTECTION EQUIPMENT

APPLICABILITY

This specification establishes the administrative requirements necessary to protect plant personnel during exposure to atmospheric concentrations of radioactive material exceeding the limits specified in Table I, Appendix B, 10 CFR 20.

OBJECTIVE

To assure, pursuant to 10 CFR 20.103(c)(3), that individuals utilizing respiratory protection equipment will not inhale excessive quantities of radioactive material.

SPECIFICATION

7.2.1 MSA CONSTANT FLOW AIR LINE RESPIRATORS WITH FULL CLEAR-VIEW FACEPIECE OR SCOTT AIR-PAK II FULL FACE PIECE AIR SUPPLIED MASKS SHALL BE USED IN ATMOSPHERES WHERE THE CONCENTRATION ANTICIPATED DURING OCCUPANCY MAY EXCEED 800 DIVIDED BY THE HOURS OF OCCUPANCY TIMES THE CONCENTRATION DETERMINED IN ACCORDANCE WITH NOTE 1 WITH RESPECT TO THE VALUES IN TABLE I, APPENDIX B, OF 10 CFR 20. FULL FACEPIECE RESPIRATORS EQUIPPED WITH FILTERS HAVING RATED EFFICIENCIES EXCEEDING 99.95% FOR 0.3 MICRON DIAMETER PARTICLES MAY BE USED IN ATMOSPHERES WHERE THE CONCENTRATIONS ARE BELOW THE VALUES SET FORTH ABOVE.

7.2.2 PRIOR TO EACH ENTRY INTO A CONTAMINATED ATMOSPHERE, INDIVIDUALS WEARING RESPIRATORY PROTECTION MASKS SHALL CHECK FOR FIT AND LEAKAGE BY BLOCKING THE FILTER AND/OR THE AIR SUPPLY LINE WHILE ATTEMPTING TO INHALE.

7.2.3 FOLLOWING EACH USE, RESPIRATORY PROTECTION MASKS SHALL BE RETURNED BY THE WEARER FOR DECONTAMINATION AND SANITIZING UNDER THE DIRECTION OF HEALTH AND SAFETY PERSONNEL. WHEN CLEANING AND REPAIR HAS BEEN APPROVED BY HEALTH AND SAFETY PERSONNEL, MASKS SHALL BE PACKAGED INDIVIDUALLY IN PLASTIC BAGS AND DELIVERED FOR ISSUE WITH CLEAN CLOTHING SUPPLIES. MASKS WILL NOT BE ASSIGNED TO INDIVIDUALS.

7.2.3.1 MASKS SHALL NOT BE RELEASED FOR RE-USE IF RADIOACTIVE CONTAMINATION EXCEEDS 200 PICOCURIES/100 CM^2 ALPHA OR 500 PICOCURIES/100 CM^2 BETA.

7.2.3.2 FILTER CANISTERS FOR MASKS SHALL NOT BE RELEASED IF THEY BEAR MORE THAN 40 PICOCURIES SMEARABLE ALPHA ACTIVITY OR RADIOACTIVE CONTAMINATION RADIATING MORE THAN 0.2 MRAD/HR AT 1 INCH AS MEASURED WITH A β-γ GM METER.

BASES

Specific criteria are called for in 10 CFR 20.103(a)(3) to establish that adequate types of respiratory protection equipment will be used in a safe manner to protect personnel when

they must work in contaminated atmospheres. The types of equipment and the procedures governing its care and use to fulfill these criteria are described in the letter from NFS to USAEC, Division of Materials Licensing dated January 25, 1965.

7.3 HIGH RADIATION AREA ACCESS

APPLICABILITY

This specification establishes the administrative procedures which are acceptable for governing entrance to a high radiation area in lieu of the alarm required by 10 CFR 20.

OBJECTIVE

To assure that each individual as well as a supervisor is aware that there is going to be an entry into a High Radiation Area.

SPECIFICATION

7.3.1 ADMITTANCE TO HIGH RADIATION AREAS DESIGNATED ZONE IV IN THE FINAL SAFETY ANALYSIS REPORT SHALL BE CONTROLLED BY LOCKED DOORS WITH THE POSSESSION OF KEYS TO THESE DOORS LIMITED TO NFS MANAGEMENT PERSONNEL.

BASES

10 CFR 20.203(c)(2) requires that a means be provided to warn personnel when they enter areas where the radiation exposure potential may be significant. This procedural specification is provided in lieu of the alarm signal device prescribed in 10 CFR 20.203(c)(2).