110741

October 12, 1966

Mr. A. E. Aikens, Jr., Chief
Irradiated Fuels Branch
Division of Material Licensing
U. S. Atomic Energy Commission
Washington, D. C.

Reference: NFS-1

Dear Mr. Aikens:

Nuclear Fuel Services, Inc., West Valley, New York hereby requests that an addition be made to 2.3 of the Technical Specifications of NFS-1 insofar as it pertains to possession limits of byproduct material. In order to properly calibrate dose rate instruments throughout the plant, NFS requests a change in the Technical Specifications to include receipt, possession and use of up to 5 millicuries of Sr^{90}. This will be purchased as $Sr^{90}...Y^{90}$ in acid solution and will be subdivided into approximately ten mounts. The individual Sr^{90} sources will be evaporated on planchets in radiochemical hoods and the entire planchet will be encapsulated in scotchcast resin or its equivalent, to prevent any loss of radioactive material. The source and holder will be quite similar to Battelle-Northwest Laboratories source holder. Radiation from the holder when not in use will be negligible as the source drops behind shielding. The prints of this source holder design are attached for your information.

Administrative controls will be imposed during encapsulation to insure against overexposure to personnel. Film badges, dosimeters and other protective equipment will be worn as necessary. The encapsulation will be performed under the direct supervision of R. B. Keely, Director of Health and Safety. Procedures employed for encapsulation will be reviewed by the members of the Plant Safety Committee prior to their implementation. A resume of Mr. Keely was included in NFS letter to Mr. A. E. Aikens dated October 29, 1963.

Very truly yours,

W. H. Lewis
General Manager
Vice President

WHL/BEK/sk
Encl: Prints

cc: J. Anderson

bcc: S. Dennis
R. B. Keely

110740

HO-16

UNITED STATES ATOMIC ENERGY COMMISSION
REGULATORY MAIL AND RECORDS SECTION

Knight

Your ☒ letter, ☐ application, dated October 12, 1966

requesting an addition to 4.3 of Tech. Specifications of CSF-1..

including enclosures thereto of drawings (2)
is acknowledged and has been assigned:

DOCKET No. 50-291 or CONTROL No.

Please refer to the above number(s) in future correspondence.

Date Received: 10-14-66

This is an acknowledgment form only.
It is not a reply to your communication.

16 1976 4 GPO

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

OCT 28 1966

IN REPLY REFER TO:
Docket No. 50-201

Change No. 1
License No. CSF-1

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Attention: Mr. W. H. Lewis
Vice President and General Manager

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

This refers to your request dated October 12, 1966, for authority to receive, possess and use up to five millicuries of strontium-90 for the preparation and use of calibration sources for dose rate instruments. Your request has been designated as Proposed Change No. 1 and has been considered pursuant to the provisions of Section 50.59 of 10 CFR 50.

We have reviewed the proposed change and have found that the proposed change does not present a significant hazard consideration not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change. Preparation of the sources will be under administrative controls according to procedures approved by the Plant Safety Committee. Film badges, dosimeters and other protective equipment will be utilized as necessary.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the following authority is added to Section 2.3 of the Technical Specifications of License No. CSF-1

"Material	Form	Possession Limit"
Strontium-90	Solution/sealed sources	5×10^{-3} curies

Sincerely yours,

(Signed) J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

cc: C. R. Braun
B. G. Bechhoefer

COPY

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

Oct 28, 1966

Docket No. 50-201

Change No. 1
License No. CSF-1

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Attention: Mr. W. H. Lewis
Vice President and General Manager

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
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We have reviewed the proposed change and have found that the proposed change does not present a significant hazard consideration not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change. Preparation of the sources will be under administrative controls according to procedures approved by the Plant Safety Committee. Film badges, dosimeters and other protective equipment will be utilized as necessary.

COPY

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Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the following authority is added to Section 2.3 of the Technical Specifications of License No. CSF-1

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Strontium-90	Solution/sealed sources	5×10^{-3} curies

Sincerely yours,

(signed) J. A. McBride)
J. A. McBride, Director
Division of Materials Licensing

cc: C. R. Braun
B. G. Bechhoefer

COPY

CHANGE NO. 2

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Export Controlled Information

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5. The pump niche drain to the PPC so leakage in the niches involves the same safety features as leakage within PPC.

This TWX will be followed with the appropriate letter for your records. We believe the foregoing change in our Technical Specifications is justified. Anything that can be done to expedite this matter will be greatly appreciated. Should any questions arise, we shall be happy to supply additional information. We wish you and your staff a very Merry Christmas.

Very truly yours,

WHL/BEK/sk

cc:

W. H. Lewis
General Manager
Vice President

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CHANGE NO. 3

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We have taken into consideration the authorization in Technical Specification 4.8.1.5 to package in specified containers the solutions that may be stored in accordance with Technical Specification 4.8.1.4. Based on data in K-643 and ORNL-2143 we have concluded that packaging and storage in accordance with Technical Specification 4.8.1.5 of solutions authorized for poisoned tank storage by this revision of Technical Specification 4.8.1.4 will be in keeping with normal safety requirements.

On the basis of our review of the nuclear safety aspects of the requested change we conclude that the proposed change does not present a significant hazard consideration not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change.

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

IN REPLY REFER TO:
Docket No. 50-201

JUN 8 1967

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Change No. 4
License No. CSF-1

Attention: Dr. R. P. Wischow
Assistant General Manager

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

This refers to (1) Nuclear Fuel Services' request dated December 16, 1966, and supplemented February 10, 1967, February 15, 1967, and March 20, 1967, for approval of a proposal to install a heat exchanger in Waste Tank 8D-2, and to use the heat exchanger to boil-off excess liquid from the waste tank, and (2) Nuclear Fuel Services' requests dated March 24, 1967, for changes in the technical specifications dealing with radioactivity in effluents, monitoring of effluents, and administrative requirements governing operation of the facility. These requests have been designated as Proposed Change No. 4, and have been considered pursuant to the provisions of Section 50.59 of 10 CFR 50.

We have reviewed the proposed installation and use of the heat exchanger in Waste Tank 8D-2 and have found that with the change in Technical Specification 4.2 authorized herewith to govern the disposal of steam condensate, the installation and use of the subject heat exchanger in Waste Tank 8D-2 does not present significant hazards considerations not described or implicit in the

NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by its installation and use. Accordingly, the proposed installation and use of the heat exchanger is authorized. A copy of the Waste Tank Heat Exchanger Safety Analysis prepared by the Division of Materials Licensing is attached.

In addition, we have reviewed the proposed change in Technical Specifications and have found that it does not present significant hazards considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change. A copy of the Evaluation of Technical Specification Change Number Four by the Division of Materials Licensing is attached.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, changes in Technical Specifications 4.0, 4.1, 4.2, 5.0, 5.1, and 7.1 of License No. CSF-1 as set forth on attached pages 11, 12, 14, 15, 16a, 16b, 16c, 43, 44, 45, 70, 71, and 72 are authorized.

Sincerely yours,

Original Signed by
J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

Attachments:
Change 4 Revision pages 11, 12, 14, 15, 16a, 16b, 16c, 43, 44, 45, 70, 71, and 72.

Evaluation of Technical Specification Change Number Four

Waste Tank Heat Exchanger Safety Analysis

cc: Mr. T. C. Runion
Mr. B. G. Bechhoefer

EVALUATION OF TECHNICAL SPECIFICATION
CHANGE NUMBER FOUR
BY
DIVISION OF MATERIALS LICENSING
NUCLEAR FUEL SERVICES, INC.
DOCKET NO. 50-201

SUMMARY

Following several discussions with regulatory personnel, Nuclear Fuel Services, Inc., (NFS) submitted two letters, each dated March 24, 1967, which proposed modifications to the Technical Specification of License CSF-1. The proposed revisions dealt with limits on the quantity of radioactivity in effluents, monitoring of effluents, and administrative requirements governing the operation of the facility. During our review of the proposed modifications we had several telephone conversations with NFS which resulted in additional modifications that were agreed upon by NFS. None of the proposed revisions presented a hazards consideration not described or implicit in the NFS "Final Safety Analysis Report". With the exception of the modification to Specification 4.1.1, described below the revisions provided for greater in-depth control of the operation. Accordingly, the revisions reasonably assured that the health and safety of the public would not be endangered and they were incorporated into License CSF-1 by our teletype of March 30, 1967. The revisions are discussed below:

EFFLUENT RADIOACTIVITY LIMITS

Specification 4.0 has been modified to correct the numbering of the technical specifications referenced therein.

Specification 4.1.1(c) has been added to allow some credit for the dilution capability provided by the meteorology of the site as is commonly authorized in the case of reactor facilities. Based on the annual average dilution data presented in the "Final Safety Analysis Report", a monthly average release rate of 0.1 microcuries per second of particulate activity has been authorized. This rate will result in concentrations at ground level off-site below the values specified in 10 CFR 20, Appendix B, Table II, Column 1 and will be consistent with Federal Radiation Council dietary intake guidance. For example,

if all of the particulate radioactivity released were the highly radiotoxic isotope Sr-90, the concentrations off-site would be 1/1000 of the allowable 10 CFR 20 concentration limit (inhalation) in unrestricted areas. In terms of soluble strontium nitrate the quantity released per year via the stack will have to be limited to 0.037 grams in order to meet the requirements of this specification.

Specification 4.1.2 has been revised to incorporate minimum values of wind velocity which shall exist before initiation of a dissolution operation. These agreed upon values will prevent dissolutions except during periods when there is reasonable assurance that gaseous radioactivity releases will not result in off-site exposure in excess of 10 CFR 20 standards.

Specification 4.2.1 has been added to provide more in-depth control of liquid effluents. The specification includes the operating technique which must be used with the single interceptor (two interceptors are to be installed in the near future), establishes a limit on the quantity of radioactivity which may be released from the interceptor to the lagoons in any quarter, and permits bypassing of the interceptor and lagoons if a liquid effluent is batchwise collected and analyses shows it to be below the concentration limits for effluent released to unrestricted areas.

Liquid effluents entering the interceptor shall be monitored by an in-line monitor which will indicate a release of radioactivity at a concentration in excess of 5×10^{-3} $\mu c/ml$ to the interceptor. The interceptor shall be sampled and the sample analyzed to determine that it is below the specified limit before the interceptor is discharged. Any alarm of the in-line monitor during discharge of the interceptor requires that the discharge operation be stopped. These operational steps should prevent an inadvertent release of radioactivity to the lagoons.

The 4.9 million gallon lagoon system provides holdup equivalent to the volume of plant waste water released during 120 days of normal operation. This holdup capability permits a period for radioactivity decay, sedimentation and ion exchange and allows release to Cattaraugus Creek during periods of high flow thus assuring that a minimum concentration will result in unrestricted areas. The maximum quantity of radioactivity that may be transferred to the storage lagoons in any quarter is equal to the quantity which could be diluted to the limits specified by 10 CFR 20 for unrestricted areas if the annual average flow rate indicated in the "Final Safety Analysis Report" has existed during the quarter.

Specification 4.2.2 is the same as the former Specification 4.2.1

EFFLUENT ACTIVITY MONITORING

Specification 5.0 has been modified to require shut-down of operations releasing radioactivity to the environment if monitoring requirements cannot be fulfilled.

Specification 5.1.1 has been added to require analytical determinations of radioactivity in all batches of effluent liquids prior to release, either to the lagoon system or directly to the environment. This guards against an inadvertent release of contaminated liquids from sources not expected to be radioactive.

Specification 5.1.2 was formerly numbered 5.1.1. In addition to the change in numbering, this specification has been made more restrictive in respect to operability of the Cattaraugus Creek continuous sampler. No more than two million gallons can be released from the lagoons in one year utilizing manual sampling in lieu of continuous sampling of Cattaraugus Creek. A minimum frequency of manual sampling has been specified. to assure that the analytical data is representative.

Specification 5.1.3 takes the place of former 5.1.2. It has been revised to require continuous collection of I-131 and particulate samples of stack effluent and weekly analyses of these samples for both alpha and beta activity of the particulates. Continuous monitoring of Kr-85, I-131 and particulate beta activity of the stack effluent is also required. If the continuous Kr-85 monitor fails, processing may continue by calculating the quantity of released Kr-85 from irradiation data of the fuel being processed, If the I-131 or particulate beta activity monitors become inoperative a sample collected throughout each shift will be analyzed.

ADMINISTRATIVE REQUIREMENTS

The scope of the administrative requirements has been increased.

In Specification 7.1.1 an Assistant General Manager position has been added with direct responsibility for radiological and nuclear safety. The Director of Health and Safety and the Technical Services Manager report to the new Assistant General Manager, and their responsibilities have been defined in terms of their radiological and nuclear safety functions. An additional change in Specification 7.1.1 adds to the Plant Safety Committee's responsibilities the investigation of abnormal occurrences and the recommendation of measures to prevent repetition of such occurrences.

Specification 7.1.3 has been revised to indicate an acceptable degree of knowledge on the part of the production operating staff.

Specification 7.1.5 has been revised to require that records be kept of Plant Safety Committee meetings as well as details of their investigations and recommendations.

Specification 7.1.6 is a new specification requiring periodic audits of radiological and nuclear safety throughout the facility by competent personnel other than those responsible for the execution of the audited operations.

Specification 7.1.7 is the former 7.1.6.

WASTE TANK HEAT EXCHANGER
SAFETY ANALYSIS
BY
THE DIVISION OF MATERIALS LICENSING
NUCLEAR FUEL SERVICES, INC.
DOCKET NO. 50-201

AUTHORIZATION REQUESTED

By a letter dated December 16, 1966 and supplemented by letters dated February 10, 1967, February 15, 1967, February 20, 1967, and March 20, 1967, Nuclear Fuel Services, Inc. (NFS) submitted design information and operating procedures for a heater to be placed in high level Waste Tank 8D-2 and requested approval of the installation and use of such a heater to boil off excess water accumulated in the tank because the radioactivity content is thus far insufficient to cause self boiling of the waste liquid. Subsequently the heater may be transferred to the spare Waste Tank 8D-1, which has been half filled with water, but to which radioactive waste has not yet been directed.

BACKGROUND

Waste Tanks 8D-1 and 8D-2 are 750,000 gallon tanks of mild steel, located underground in concrete vaults. They are designed to contain neutralized self boiling radioactive wastes from plant operations. The tanks are provided with vent systems that include condensers, a knock-out drum, reheater, filters, and exhausters. Uncondensed gases are discharged to the plant ventilation system. Condensate may be returned by gravity to the waste tanks or, if reduction of tank contents is desired, it may be pumped to the plant for processing and discharge to the environment.

Prior to the introduction of neutralized radioactive waste to Tank 8D-2, 100,000 gallons of caustic solution was added to the tank to assure neutralization and an additional 300,000 gallons of water was added to assure proper operation of built-in air lift circulators. Because of these water additions and the low heat generation rate of the radioactive waste being transferred to Tank 8D-2, the tank will not boil by itself before it becomes filled. Hence, the need to install a heat exchanger to boil off the excess water.

The proposed installation will consist of a steam heated tube bundle vertically suspended in the tank from a concrete filled plug mounted in the 12-inch corrosion coupon access hatch of the tank. Corrosion specimens in 8D-1 and 8D-2 now attached to the 12-inch hatch covers will be moved at the time of the heat exchanger installation to the 6-inch sampling hatch covers. Future sampling of the tanks will be effected through a 2-inch hole in the heat exchanger installation plug and a hole in a new 12-inch plug to be used after the heat exchanger is removed. Thus, the proposed installation will not interfere with normal corrosion monitoring and sampling programs of the waste tanks.

A 2-inch steam line and a 1-inch condensate line, cast in the hatch plug, will feed and drain the heat exchanger. Steam will be supplied via a temporary 2-inch line mounted above ground. Steam flow will be regulated by using a globe valve and pressure gauge. A bleed valve on the steam line will permit venting the exchanger system upon termination of steam flow to relieve the vacuum caused by condensing steam. Condensate from the heat exchanger will pass through a steam trap for batchwise collection in a portable 3500 gallon tank, where it will be sampled and conveyed to appropriate disposal. The condensate line is also provided with a sample tap at the steam trap and with piping and valves for routing the condensate directly to Tank 8D-2 rather than to the portable collection tank.

SAFETY CONSIDERATIONS

1. Control of Activity During Installation and Removal of the Exchanger

The NFS procedures for installing the heat exchanger call for removing the present corrosion coupon access hatch cover with a crane and then encapsulating the cover for storage. The new plug and heat exchanger will be inserted into the hatch immediately. If any delay is encountered, the hatch will be recovered with the original plug or, if the delay is only a few minutes, a temporary plastic cover will be used. Once the heat exchanger is in place the tank will be in a sealed condition. When transferring the heat exchanger to Tank 8D-1, the heat exchanger will be slowly withdrawn from Tank 8D-2 into a plastic bag as it is being sprayed down and rinsed with the rinse going back into Tank 8D-2. The hatch will be capped at once with the plug from the corrosion coupon access hatch on 8D-1 and the heat exchanger and plug will be transported by crane to the 8D-1 corrosion coupon access hatch where it will be inserted into Tank 8D-1. In addition to these procedures, NFS notes that the off-gas system exhauster will provide a draft to insure that the flow of air will be into the hatch and not out. NFS further notes that they have experienced no venting of tank vapors to the atmosphere when sampling the tanks through the 6-inch hatch.

2. Venting Capacity and Entrainment

Approximately 3000 pounds per hour of condensate will be discharged from the heat exchanger (2.7×10^6 BTU/hr) when the tank is boiling. About ten percent of the heat will be lost to the ground around the tank, thus the maximum net boil-off rate from the tank will be 2700 pounds per hour. This quantity is within the design rating of the condensers in the tank vent system.

NFS estimates that the vapor velocity at the vapor exit line from the tank will be only 2/5 the design value. Also, the fission product concentration is only about 1/10 of the concentration that will be present when the tank is ultimately filled to the design level with high activity waste. Under these conditions NFS sees no problem with entrainment of activity into the vent system.

Since these tanks were designed for higher boiling rates, we believe that the proposed operation will present no venting or entrainment hazards not previously analyzed.

3. Leakage of Radioactivity Into Heater System

During routine operation, the heat exchanger will be at a minimum pressure of 15 psi and no activity would be expected to leak into the heat exchanger should it develop a leak during use. The condensate from the heat exchanger will be collected in a portable 3500 gallon tank, sampled, and then, if the activity concentration is within specified limit, it may be released to the environment. If the condensate becomes contaminated above the release limit, it will be transferred to either Tank 8D-2 or the General Purpose Evaporator, 7C-5.

To assure that this method of condensate disposal is adequate, Technical Specification 4.2.1 incorporates requirements for batch collection and sampling of condensates and specifies that such condensate may not be released directly to the environment if a concentration of 0.3 picocuries per milliliter is exceeded.

Whenever steam flow to the heat exchanger is terminated, the steam trap will be bypassed to relieve the steam pressure, then the bleed valve will be opened to relieve the vacuum caused by condensation of steam in the system. When the system is restarted, the valving will be adjusted to close the vents to atmosphere. A steam pressure of 20 psig will be built up with the heater condensate routed to the high level waste tank. After a sample of the flowing condensate has yielded a satisfactory activity analysis, the heater condensate will be routed to the portable collection tank.

These precautions, along with the sampling of the condensate collected in the portable tank provide ample surveillance to warn of leaks which might develop.

4. Piping Failure

NFS presented final design details which included a description of features to reduce piping stresses. The staff reviewed the design and concurs that the design provides adequate strength to make mechanical stress failures unlikely.

If a complete break should occur in the 2-inch steam line, NFS estimates that approximately 4000 lbs/hr of steam or about 4.9×10^6 BTU/hr, could be released into the waste tank. We have independently verified that the flow would be so constrained by the characteristics of the piping. Since the tank vent system condenser is designed to handle 1×10^7 BTU/hr, the maximum steam leak rate would be accommodated, and hence this does not present a hazard not previously analyzed.

CONCLUSION

Based on our review of the proposed installation, we conclude that it does not present significant hazards considerations not described or implicit in NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed heat exchanger in the high level waste tanks. We recommend that the heater steam condensate discharged to the environment be limited to a radioactivity concentration that does not exceed the unrestricted area concentrations specified in 10 CFR 20.

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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 to 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.

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

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^{-6} 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.

The limit for particulate release is based on the normal dilution reducing the average concentration to 2×10^{-14} microcuries per milliliter or less at ground level beyond the exclusion area.

(Change No. 4 Revision)

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 ALL PROCESS STEAM CONDENSATE SHALL BE COLLECTED BATCHWISE AND SAMPLED FOR GROSS ALPHA AND BETA ACTIVITY. THE CONDENSATE MAY BE DISCHARGED WITHOUT PASSING THROUGH THE STORAGE LAGOONS AFTER IT IS DETERMINED THAT ITS CONCENTRATION IS 0.3 PICOCURIES PER MILLILITER OR LESS. ALL OTHER LIQUID PLANT EFFLUENTS CONTAINING RADIOACTIVITY FROM PLANT OPERATION OR WHICH HAVE BEEN IN CONTACT WITH EQUIPMENT CONTAINING RADIOACTIVITY SHALL BE MONITORED BY AN INLINE MONITOR SET TO ALARM AT 5 NANOCURIES PER MILLILITER AND SHALL BE COLLECTED IN THE INTERCEPTOR. THE INTERCEPTOR DISCHARGE VALVE SHALL BE FULLY OPENED AFTER THE INTERCEPTOR HAS BEEN SAMPLED AND THE ANALYSIS VERIFIED TO BE LESS THAN 5 NANOCURIES PER MILLILITER. THE CONTENTS SHALL BE DISCHARGED AS RAPIDLY AS POSSIBLE. WHEN THE INTERCEPTOR IS EMPTY, THE DISCHARGE VALVE SHALL BE CLOSED AND THE CYCLE REPEATED. IF THE INLINE MONITOR ALARMS DURING DISCHARGE OF THE INTERCEPTOR, THE DISCHARGE VALVE SHALL BE CLOSED IMMEDIATELY AND THE CAUSE OF THE DIFFICULTY DETERMINED AND CORRECTED. DURING ANY CALENDAR QUARTER NO MORE THAN 48 CURIES COMBINED EQUIVALENT UNIDENTIFIED ALPHA AND BETA ACTIVITY OTHER THAN TRITIUM OR I-131 SHALL BE TRANSFERRED TO THE LAGOONS.

4.2.2 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

These specifications limit the radioactivity in the storage lagoons and the radioactivity concentration in Cattaraugus Creek. The discharge of liquids from the site has been discussed in paragraph 7.11 through 7.13 of the Final Safety Analysis Report. The dominant isotope in the liquid effluent is expected to be tritium for which a separate analysis will be made. (See Specification 5.1.2). In addition, the quality of liquid effluents may be such that the quantity of other isotopes such as Sr-89, Sr-90, Cs-137, Ce-144, Ru-106, I-129, I-131, and Zr-Nb-95 may also have to be specifically measured.

The alarm setting of the inline monitor will be established in terms of beta activity other than tritium as determined by analyses of the liquid flowing through the neutralization tank As shown below, the limited plant throughput of tritium makes it unnecessary to detect tritium in analyses of liquid samples from the interceptor.

"Combined equivalent unidentified curies" will be calculated as follows.

When specific analyses are made for any isotope, the curies of that isotope found will be multiplied by the ratio of $\frac{3\times10^{-7}}{MPC_i}$.*

The "equivalent curies" so determined will be summed as illustrated in the following example to determine the total amount released to the lagoons. I-131 is a special case. For I-131 a minimum holdup time of 48 days (6 half lives) is assumed in the lagoon system, and credit is taken for this decay.

EXAMPLE:

Assume that an interceptor batch is found to contain 10 curies of radioactivity other than tritium, plus 600 curies of tritium (from 2 tonnes of fuel). Specific analyses are made which show that the isotopic composition, except for tritium, is as follows:

Ru-106	40%
Zr-95	30%
I-131	25%
Other (unidentified)	5%
Total	100%

*3×10^{-7} microcuries per milliliter is the maximum permissible concentration in unrestricted areas for Sr-90 which is one of the most radiotoxic isotope expected to be released in the liquid effluent from an irradiated nuclear fuel processing plant. MPC_i is the maximum permissible concentration for each particular isotope as indicated in Appendix B, Table II, Column 2 of 10 CFR 20.

(Change No. 4 Revision)

Then the "equivalent unidentified curies" are:

Isotope	Measured Curies	MPC_i	$\frac{3x10^{-7}}{MPC_i}$	Iodine Factor	Equivalent Unidentified Curies
Ru-106	4	$1x10^{-5}$	0.03	-	0.12
Zr-95	3	$6x10^{-5}$	0.0048	-	0.015
I-131	2.5	$3x10^{-7}$	1	0.0155	0.04
Other (unidentified)	0.5	$3x10^{-7}$	1	-	0.5
		Combined Equivalent Unidentified Curies			0.675

The contribution of the tritium would be calculated as follows:

Isotope	Measured Curies	MPC_i	$\frac{3x10^{-7}}{MPC_i}$	Iodine Factor	Equivalent Unidentified Curies
H-3	600	$3x10^{-3}$	$1x10^{-4}$	-	0.06

It is apparent from these calculations that tritium need not be specifically analyzed to determine its contribution to the "combined equivalent unidentified curies."

It is not the intent of NFS to discharge from the plant the maximum quantity of radioactivity which could be accommodated within permissible concentrations by the flow of Cattaraugus Creek. Prudent plant operation dictates that the quantity of radioactivity released to Cattaraugus Creek be kept at the lowest practical amount. To assure an in-depth control of liquid effluents released to Cattaraugus Creek, a quarterly limit of 48 curie equivalents of liquid discharged to the lagoons has been set, and the concentration of liquid fed to the lagoons is limited to $5.x10^{-3}$ $\mu c/ml$. These limits are consistent with meeting the requirements of Specification 4.2.2 in Cattaraugus Creek.

Activity released into the liquid effluent system goes through 4 separate holdup units, via an interceptor, a 300,000 gallon lagoon, and two successive 2,300,000 gallon lagoons. Thus, there are four separate chances to apprehend errant activity and the total possible holdup time is about four 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 if this Technical Specification cannot be met.

(Change No. 4 Revision)

Continuous sampling will be conducted at the monitoring station near the site boundary on Cattaraugus Creek. At least on a weekly basis, the composite sample from this station (see Specification 5.1.2) will be analyzed for gross alpha and gross beta activity and for tritium. 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.

An auxiliary monitoring station has been provided near the plant site at the confluence of Frank's and Quarry Creeks. This station serves as a sensitive monitor of any unsuspected or unregulated discharges. Measurement of flow at this point and maintenance of the system is quite difficult due to the rugged terrain. Consequently data taken here cannot be expected to have the consistency necessary to make use of this station as a part of the discharge inventory. It will, however, be maintained in operation to the greatest extent possible as an added margin of safety.

(Change No. 4 Revision)

5.0 MINIMUM CONDITIONS FOR OPERATION

The specifications included in this section set forth minimum conditions for safe plant operation. If Specifications 5.1.1 through 5.1.3 cannot be fulfilled the operations which could cause a release of radioactive effluents must be shutdown with the exception of the ventilation system which normally is kept on, until the problems are corrected. The other specifications in this section provide primary and alternate conditions which may be fulfilled. If the alternate condition is in use the operations shall be considered to be continuing in a "ready condition." If neither primary nor alternate conditions for a particular operation can be fulfilled the operations shall be shutdown with the exception of the waste tank off-gas system which has to remain on stream while appropriate repairs are made.

5.1 EFFLUENT MONITORING

APPLICABILITY

This specification applies to the equipment necessary, the frequency of sampling, and the analyses required for radioactive gaseous and liquid plant effluents.

OBJECTIVE

To assure that monitoring equipment will be in operation and sampling and analyses will be performed to permit appraisal of liquid and gaseous radioactive effluents.

SPECIFICATION

5.1.1 ALL BATCHES OF LIQUID PLANT EFFLUENTS COLLECTED IN ACCORDANCE WITH SPECIFICATION 4.2.1 SHALL BE ANALYZED FOR RADIOACTIVITY CONTENT.

5.1.2 A CONTINUOUS SAMPLER WHICH OBTAINS A REPRESENTATIVE SAMPLE FROM CATTARAUGUS CREEK SHALL BE IN OPERATION WHENEVER LIQUID IS DISCHARGED FROM THE STORAGE LAGOONS TO VERIFY CONFORMANCE WITH THE RADIOACTIVE CONCENTRATION LIMITS ESTABLISHED BY SPECIFICATION 4.2.2. AT LEAST A WEEKLY COMPOSITE SAMPLE FROM THIS STATION SHALL BE ANALYZED FOR TRITIUM AND FOR GROSS ALPHA AND GROSS BETA ACTIVITY. IF THE CONTINUOUS SAMPLER FAILS, A MAXIMUM OF 2,000,000 GALLONS MAY BE DISCHARGED ANNUALLY PROVIDED THAT SAMPLES OF CATTARAUGUS CREEK ARE COLLECTED MANUALLY TO VERIFY CONFORMANCE WITH SPECIFICATION 4.2.2. IF MANUAL SAMPLING IS NECESSARY, THE FREQUENCY SHALL BE AT LEAST ONE SAMPLE PER DISCHARGE OR PER 100,000 GALLONS DISCHARGED, WHICHEVER RESULTS IN THE GREATER SAMPLING FREQUENCY.

5.1.3 THE I-131 AND PARTICULATE SAMPLER LOCATED BESIDE THE STACK WHICH COLLECTS A REPRESENTATIVE SAMPLE OF THE STACK GAS SHALL BE IN OPERATION AT ALL TIMES AND THE SAMPLES COLLECTED SHALL BE ANALYZED AT LEAST WEEKLY TO VERIFY CONFORMANCE WITH SPECIFICATION 4.1.1. IN ADDITION, THERE SHALL BE CONTINUOUS MONITORING OF THE STACK GAS FOR I-131, KR-85, AND PARTICULATE RADIOACTIVITY. IF THE CONTINUOUS MONITOR FAILS, A SAMPLE COLLECTED THROUGHOUT EACH SHIFT SHALL BE ANALYZED FOR I-131 AND PARTICULATE RADIOACTIVITY; KR-85 MAY BE CALCULATED FROM FUEL BURNUP DATA.

BASES

Technical Specifications 4.1 and 4.2 establish quantity or concentration limits for radioactivity released from the plant into the atmosphere and the watercourse. This specification establishes the locations where monitoring shall be performed and the monitoring equipment necessary to show with a high degree of certainty that compliance with Specifications 4.1 and 4.2 is attained.

The methods of operating the low level radioactive liquid waste discharge systems are discussed in Sections 7.11 to 7.13 of the Final Safety Analysis Report and in Section XIV paragraph A5 of Submission No. 27 of the Final Safety Analysis Report. These sections indicate that low-level radioactive liquid waste will be collected batchwise in interceptors at an approximate rate of 40,000 gallons per day and will be sampled and analyzed. The weekly samples at the off-site monitoring station will be analyzed for tritium and gross alpha and beta activity. This modus operandi should provide knowledge of the quantity of radioactivity released to the storage lagoons and released to Cattaraugus Creek and will support a showing of compliance with Specification 4.2.

The weekly analyses of the composite samples collected of radioiodine and radioactive particulates will provide a basis for determining the compliance with Specification 4.1.1.

(Page 46 is deleted)

(Change No. 4 Revision)

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 INCLUDING, BUT NOT LIMITED TO, THE FOLLOWING:

7.1.1.1 A GENERAL MANAGER, HAVING THE OVERALL RESPONSIBILITIES 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 AN ASSISTANT GENERAL MANAGER, REPORTING TO THE GENERAL MANAGER, DIRECTLY RESPONSIBLE FOR ALL THE ACTIVITIES AT THE PLANT INVOLVING HEALTH AND SAFETY, NUCLEAR SAFETY, AND ADHERENCE TO THE LIMITS AND CONDITIONS SET FORTH IN THIS LICENSE. THE HEALTH AND SAFETY DIRECTOR AND THE TECHNICAL SERVICES MANAGER REPORT TO THIS ASSISTANT GENERAL MANAGER.

7.1.1.3 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.

7.1.1.4 A HEALTH AND SAFETY DIRECTOR, REPORTING TO AN ASSISTANT GENERAL MANAGER, AND RESPONSIBLE FOR MONITORING THE RADIOLOGICAL SAFETY OF ALL PLANT ACTIVITIES AND FOR ADVISING ALL DEPARTMENTS ON RADIOLOGICAL SAFETY MATTERS.

7.1.1.5 A TECHNICAL SERVICES MANAGER, REPORTING TO AN ASSISTANT GENERAL MANAGER, AND RESPONSIBLE FOR REVIEW OF PROPOSED CHANGES IN DESIGN WITH RESPECT TO RADIOLOGICAL AND NUCLEAR SAFETY, AND RESPONSIBLE FOR A CONTINUOUS REVIEW OF OPERATIONS TO ASSURE NUCLEAR SAFETY.

7.1.1.6 A PLANT SAFETY COMMITTEE RESPONSIBLE FOR: REVIEW AND APPROVAL OF ALL STANDARD OPERATING PROCEDURES AND LETTERS OF AUTHORIZATION; REVIEW AND APPROVAL OF ALL CHANGES IN THE PROCESS, THE PROCESS SYSTEM, AND STANDARD OPERATING PROCEDURES; INVESTIGATION OF ABNORMAL OCCURRENCES WHICH MAY AFFECT RADIOLOGICAL SAFETY OR CRITICALITY PREVENTION; AND RECOMMENDATION OF MEASURES WHICH WILL PREVENT REPETITION OF SUCH ABNORMAL OCCURRENCES. THE COMMITTEE SHALL CONSIST OF, BUT NOT BE LIMITED TO, THE GENERAL MANAGER, THE ASSISTANT GENERAL MANAGER, THE TECHNICAL SERVICES MANAGER, THE HEALTH AND SAFETY DIRECTOR, AND THE PRODUCTION MANAGER.

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

7.1.2.1 PREPARATION, APPROVAL, AND ISSUANCE OF ALL OPERATING INSTRUCTIONS AND CHANGES THERETO, INCLUDING, BUT NOT BE LIMITED TO: STANDARD OPERATING PROCEDURES, LETTERS OF AUTHORIZATION, RUNSHEETS, 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 BE 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. MEMBERS OF THE PRODUCTION OPERATING STAFF SHALL HAVE A THOROUGH KNOWLEDGE OF THE PROCEDURES GOVERNING THE WORK FOR WHICH THEY ARE RESPONSIBLE. DRILLS SHALL BE HELD AT SUFFICIENT FREQUENCY TO ENSURE PROFICIENCY IN EMERGENCY PROCEDURES. 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 HANDLING ABNORMAL OPERATING CONDITIONS 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 INCLUDE MINUTES OF MEETINGS OF THE PLANT SAFETY COMMITTEE AS WELL AS DETAILS OF INVESTIGATIONS AND RECOMMENDATIONS IN REGARD TO ABNORMAL OCCURRENCES AND ACTIONS TAKEN THEREON.

7.1.6 AUDIT OF OPERATIONS. MANAGEMENT SHALL PROVIDE FOR PERIODIC AUDITS OF THE OPERATIONS THROUGHOUT THE ENTIRE FACILITY WITH REGARD TO RADIOLOGICAL AND NUCLEAR SAFETY. THESE AUDITS SHALL BE PERFORMED BY COMPETENT COMPANY OR OTHER TECHNICAL PERSONNEL NOT DIRECTLY RESPONSIBLE FOR THE OPERATION AUDITED.

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

7.1.7.1 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.

7.1.7.2 IMMEDIATE STEPS SHALL BE TAKEN TO ESTABLISH THE CAUSE OF THE OCCURRENCE IN 7.1.7.1 BY RESPONSIBLE MANAGEMENT REVIEW.

7.1.7.3 NOTWITHSTANDING ANY OTHER PROVISION OF THIS LICENSE THE PLANT SHALL BE SHUTDOWN AND NORMAL PROCESSING OPERATIONS MAY NOT BE RESUMED PRIOR TO AEC REGULATORY CONCURRENCE WHEN SPECIFICATION 4.1.1 OR 4.2.2 HAS BEEN EXCEEDED.

(Change No. 4 Revision)

January 24, 1967

Mr. [illegible] Aikens, [illegible]
Irradiated Fuels Branch
Division of Material Licensing
U. S. Atomic Energy Commission
Washington, D. C.

Dear Mr. Aikens:

Nuclear Fuel Services, Inc. West Valley, New York hereby requests that Technical Specification 7.3 of [illegible]-1 be changed to read as follows:

SPECIFICATION

7.3.1 Admittance to high radiation areas where an individual might receive a dose of 100 millirem in one hour shall be controlled by locked doors with the possession of keys to these doors limited to NFS management personnel.

7.3.2 In lieu of locked doors for high radiation areas, where doors cannot readily be locked, a system such as stanchings connected by ropes with high radiation signs attached to all sides will be used so that the radiation level at the boundary will be 5 mr/hr or less.

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. The above procedure complies with the intent of 10 CFR 20. Some areas of the plant such as staircases that are necessary for evacuation purposes have high radiation levels from time to time. Since these areas cannot be readily locked and for safety reasons should not be locked out, we propose to use the method described in 7.3.2 to warn personnel of high radiation areas, so that they can not unintentionally enter these areas.

We believe the above change in the Technical Specifications is justified. Should any questions arise, we shall be happy to supply additional information.

Very truly yours,

W. H. Lewis
General Manager
Vice President

WHL/BEK/ak

cc: J. Anderson

bcc: M. E. Weech
[illegible]

CHANGE NO. 5

NFS

NUCLEAR FUEL SERVICES, INC.

A SUBSIDIARY OF W. R. GRACE & CO.

BOX 124 ··· WEST VALLEY, N.Y. 14171

AREA CODE 716 TELEPHONE 942-3235

110898

September 21, 1967

Dr. John A. McBride, Director
Materials Licensing Division
U. S. Atomic Energy Commission
Washington 25, D. C.

Dear Dr. McBride:

Nuclear Fuel Services, Inc. hereby requests a change to Technical Specification 2.3 of Provisional Operating License No. CSF-1. We wish to increase the possession limit for Cesium-137 so that we may use a sealed source for calibrating survey instruments. We propose to obtain a double encapsulated, nominal 30 curie source (U. S. Nuclear Corp. Model CCSD-30 M).

We request that Specification 2.3 be changed to include possession of 33 curies of Cesium-137 as a sealed source. This source has been discussed with Mr. W. Ray of your office. If further information is necessary, please contact me.

Very truly yours,

J. P. Duckworth, Manager
Technical Services

JPD:JRC:ah

cc: T. C. Runion
B. G. Bechhoefer
J. D. Anderson

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

IN REPLY REFER TO:

Docket No. 50-201

OCT 16 1967

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Change No. 5
License No. CSF-1

Attention: Mr. W. H. Lewis
Vice President and
General Manager

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

This refers to Nuclear Fuel Services' request dated September 21, 1967, for a change to be made in Technical Specification 2.3 to authorize possession of 33 curies of cesium 137 as a sealed source for calibrating survey instruments. This request has been designated as Proposed Change No. 5 and has been considered pursuant to the provisions of Section 50.59 of 10 CFR Part 50, and of 10 CFR Part 30.

We have reviewed the proposed change and have found that it does not present significant hazard considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change. Technical Specification 2.3 currently authorizes the possession and use of materials for calibrating survey instruments. Authorization to possess and use cesium 137 would provide NFS with a greater flexibility in calibrating instruments. The cesium 137 will be in a sealed source assuring confinement.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, and the requirements of 10 CFR Part 30, the following authority is added to Section 2.3 of the Technical Specifications of License No. CSF-1:

"Material	Form	Possession Limit"
Cesium-137	Sealed Source	33 curies

Sincerely yours,

Original Signed by
J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

cc: Mr. T. C. Runion
Mr. B. G. Eschhoefer
Mr. Jon D. Anderson

110725

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

JAN 27 1967

IN REPLY REFER TO:
Docket No. 50-201

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Attention: Mr. W. H. Lewis
Vice President and General Manager

Gentlemen:

In your application dated December 16, 1966, you requested approval to install a heat exchanger in Waste Tank 8D-2 to boil off excess water. Since the proposed installation involves the possibility of a nuclear accident of a different type than those previously analyzed, please submit a safety analysis report of the installation and its operation in accordance with §50.59(d) of 10 CFR 50.

The safety analysis should include, as a minimum, the following:

1. A complete description of the proposed installation including a description of the piping and controls for providing steam to and removing condensate from the heat exchanger.

2. A description of the methods of controlling the dispersion of radioactivity during installation and servicing of the heat exchanger.

3. An evaluation of the effects of mechanical stresses on the system due to differential expansion of piping, buoyancy effects, and the movements induced by the convection currents in the boiling fluid.

4. An evaluation of the consequences of and a description of the means for coping with the development of leaks in the heat exchanger and piping. Consideration should be directed to but not limited to the following:

   a. Suck back of radioactive waste solutions into the heat exchanger and piping upon the termination of steam flow.

b. Discharge of radioactive waste solutions with the condensate from the heat exchanger.

c. Overloading of the waste tank condenser if a leak develops in the heat exchanger or its associated piping in the tank. Our calculations indicate that a complete break in the two-inch Schedule 40 inlet steam pipe could release sufficient steam to overload the on-stream waste tank condenser.

d. Dispersion and entrainment of activity in the steam, resulting in a carry-over of radioactive material to the waste tank condensers, if a break occurs in the heat exchanger or its piping just below the liquid surface.

Sincerely yours,

A. E. Aikens, Jr.

A. E. Aikens, Jr., Chief
Irradiated Fuels Branch
Division of Materials Licensing

cc: Mr. Oliver Townsend

110727

Heat Exchanger Data Sheet

Service Waste Tank Heater — Item No. BE-3

Manufacturer Type — Type Tube bundle in kettle — Size 8" OD x 12'

No. of Units — Trans. Per Shell 75.4 sq ft

Tube Sheets — Area Total 75.4 sq ft

Performance

	Shell Side	Tube Side
Fluid	Alkaline Salt Solution	SAT. STM. 75 PSIG
Total lb/hr	510,000	2 000
Liquid	10,000	—
Vapor		
Non-cond.		
Steam		2000
Fluid Vap. or Cond.	1900	2000
Sp G Liquid	1.40 @ 320°F	1.0
Visc Liquid CP	1.8 @ 320°F	
Mol. Wt. vapors	18	18
Temp. in °F	320	320
Temp. out °F	320	320
Press in PSIG	5	75
No of Passes	Kettle	1
Vel. Ft/Sec		
Press Drop Clean PSI		
Fouling Factor Req'd.	0.001	0.0005

Heat Exchanged BTU/Hr 1,855,000 — Mean Temp Diff 100 °F

Transfer Rate Service 250 — Clean 400 — Fouling Factor 0.0015

Construction

Design Press PSIG	—	150
Test Press PSIG	—	225
Design Temp °F		375

Tubes No. 30 — OD in. 3/4 x 12 BWG — Length ft. 12 — Pitch 1" Square

Shell NONE in. id.

Connections Shell in — out —

Channel in 2" Steel Socket weld out 1" Steel Socket weld

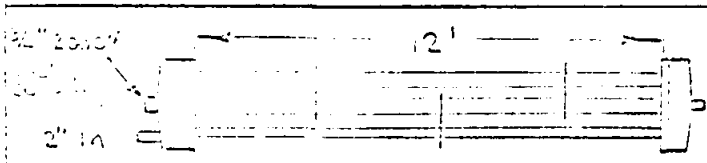
Baffles 3 - 40% Cut Segmental

Code Requirements NONE — ASME — TEMA

wtg lbs each shell & bundle — Full of water 510 — Bundle 450

Part	Material	thk. in	Corr. all.	Part	Material	Thk. in	Corr. all.
Tubes	SA179 Steel	12 BWG	0	Tube Sheets	SA285 Steel	1 1/8"	0
Shell	NONE						
Shell Cover	NONE						
Channel	STEEL	1/2"	0				
Channel Cov.	STEEL	1 1/8"	0				
Baffles	STEEL	1/8"	0				

Remarks: 1. Tubes to be Seal welded to Tube Sheets
2. Bundle to be used in vertical position
3. Channels and covers to be all welded and welded to tube sheets
4. Entire bundle must pass through 10 inch I.D. opening



Date	Revisions		
11/4/66	ISSUED FOR PURCHASE	1	[illegible]
11/1/66	ISSUED FOR QUOTE	0	[illegible]

110728

Knight

February 10, 1967

United States Atomic Energy Commission
Division of Materials Licensing
Washington, D. C.

Attention: Mr. [illegible] Hirons, Jr.
Chief, Irradiated Fuels Branch

Dear Sirs:

Your letter of January 27, 1967 asked a number of questions on our proposal to boil down the contents of Waste Tank, 8C-2, using a heat exchanger installed through one of the tank nozzles. Our answers to these questions are enclosed.

1. Description of Installation and Operation

The waste tank (8C-2) is now at 72°F and contains approximately 460,000 gallons of liquid. For each degree of temperature rise 3950 pounds of steam will have to be supplied to the heat exchanger. The heat exchanger and steam piping will not deliver and condense more than 3300 pounds of steam per hour at the start of heating which will limit the temperature rise in 8C-2 to approximately 20° per day. In actual practice the temperature rise will be kept below the allowable 20° per day to allow the system to warm gradually and avoid differential heating and stressing of the tank and vault. When the tank contents have reached boiling (approximately 220°F) the heat exchanger and steam line will deliver no more than 3000 pounds condensate per hour. Since, approximately, ten percent of this heat will be lost to the ground around the tank, maximum net boil off from 8C-2 will be 2700 pounds of water per hour. This quantity will be within the capacity of 7C-5 for re-evaporation.

The installation will consist of a steam heated tube bundle suspended vertically through the 12" nozzle designated as the corrosion hatch on 8C-2. The bundle will be 12' long and will reach to within 2' of the bottom of 8C-2. The bundle will be suspended from a concrete filled plug mounted at the top of the corrosion hatch and will be serviced by a 2" steam line and 1" condensate line cast in the plug. Steam supply will come from the area of the waste tank equipment enclosure through approximately 150' of 2" Sch-40 temporary pipe mounted above ground. Condensate will be returned through a temporary 1" line approximately 30 feet to existing 8P25-2"A at the discharge of pump 8C-5. This connection will allow steam condensate to flow to the waste lagoons but there will also

be a by-pass to conduct the steam condensate into 8D-2. Steam flow will be regulated by means of a 2" globe valve and pressure gage. There will be a sample point at the steam trap to permit condensate sampling. Details are shown on drawings 8D-F-277-3, 4413-15A-A-102-4, 4413-8A-L-2-8 and heat exchanger data sheet 8L3.

2. Activity Control During Installation and Removal of Exchanger

During installation of the heat exchanger, the plug on the 12" hatch will be out for a period of several minutes; however, the waste tank off-gas blower system, 8K-1 or 8K-1A will provide sufficient draft to insure that air flow will be into the hatch and not out. This air will then pass through the off-gas filters, 8F-1 and 8F-1A, before going to the stack. No difficulty with venting of tank vapors to atmosphere has been experienced in removing the 6" sampling hatch in this manner when waste tank samples are taken.

The present corrosion hatch will be removed with a crane, encapsulated, and stored. The new plug and heat exchanger assembly will be inserted into the hatch immediately. If any delay is encountered the hatch will be re-covered with the original plug or if the delay is only a few minutes a temporary plastic cover will be used. Once the heat exchanger is in place the tank will be in a sealed condition.

No servicing of the heat exchanger is anticipated after it is in place. If it should develop a leak it would be removed, decontaminated and replaced with a new one if required. At the end of the useful life of the heat exchanger it will be removed by slowly withdrawing it into a plastic bag as it is being sprayed down and rinsed with rinse going back to 8D-2. The hatch will be capped at once with the plug from the corrosion hatch on 8D-1 and the heat exchanger and plug assembly will be transported by crane to 8D-1 corrosion hatch where it will be inserted in 8D-1. Corrosion specimens in 8D-1 and 8D-2 now attached to the 12" hatch covers will be moved at the time of the heat exchanger installation to the 6" hatch covers. Future sampling of 8D-1 and 8D-2 will be effected through 2" holes in the new heat exchanger plug and a new plug to be made for the 3D-1 12" hatch. This will make it possible to sample without removing the 6" plugs.

3. Mechanical Stresses and Movement of the Exchanger and Piping

The greatest bending stresses occur when the steam is just admitted into the system and the steam inlet pipe is hot and the condensate line is still cold. The maximum stress due to deflection occurs in the steam inlet pipe and reaches 10,300 psi during this transient condition.

The total bending and load stresses on the steam condensate line approach 62,000 psi at the point where the condensate line is welded to the exchanger head. This is, of course, too high. A gusset will be welded between the condensate line and the exchanger head to reduce this stress

below the 20,000 psi limit. This represents a change in the design originally sent you. This change is shown on drawing 8C-T-377, Rev. 3. The stress calculations are shown on pages 1 through 5 of the attached design calculations.

The exchanger-piping system is overdamped and as such will not oscillate from side thrusts on the system. The exchanger and piping will deflect from such a thrust but will return to its original position exponentially with time.

The greatest side thrust, exempting movement due to temperature effects, would result from a rupture of the steam condensate line in the horizontal run just below the lower end of the exchanger. This thrust, of approximately 9 pounds, would deflect the bottom of the exchanger by approximately 10 inches. This is less deflection than the temperature expansions cause so consequently the stresses are lower than those from temperature effects. The effects of steam thrust, assuming all steam being boiled off leaves from one side of the bundle, the thrust from buoyance, and from intank circulation currents were found to be negligible.

System oscillation and hydraulic thrust calculations are given on pages 6 through 9 of the design calculations.

4. <u>Leaks in the system</u>

a) Upon termination of steam flow to the heat exchanger the trap will be by-passed and the two inch shut off valve at the beginning of the temporary line will be closed. After venting for five minutes through the trap by-pass the small bleed valve at the pressure gage will be opened and the exchanger will be left in this condition until it is to be restarted.

Upon restarting, the trap by-pass, inlet vent, and trap block valves will first be closed and the steam turned on to the heat exchanger. When a pressure of 20 psig is established the 2" ball valve will be partially opened and after condensate has circulated for several minutes a sample will be taken from the sample valve. This sample will be checked in the lab for radioactivity and if found to be below release limits to the lagoon, the ball valve will be closed and the trap valves opened, allowing condensate flow to the lagoon. If activity above release limit is found to persist procedure outlined in 4 b) will be followed.

b) During routine operation, the condensate will not be sampled since a minimum pressure of 15 psi will be maintained in the exchanger tubes. Persistent activity in the condensate during start-up will indicate an exchanger leak. If a major leak has developed, repair can only be effected by complete replacement of the heat exchanger. If the leak is minor and the exchanger is still serviceable a new line will be required to return all steam condensate back to the suction of pump 8G-1 in the equipment enclosure so that it can be returned along with the process condensate to the general purpose evaporator (7C-5) in the acid recovery cell.

c) A complete break in the 2" steam supply line will release approximately 4000 lbs/hr. of steam inside 8D-2. The 8D-2 condenser system is designed to take 10×10^6 lbs/hr. or 2.5 times this maximum leak rate. Calculation of steam flow under these conditions is given on pages 9 and 10 of the design calculations.

d) The vapor velocity at the vapor exit line from the tank will reach 1/2.5 the values reached when the tank reaches full boil-up rates. In addition the heater is located 15 ft. horizontally and approximately 12 ft. vertically from the vapor exit nozzle. The vapor will traverse a total of approximately 19 feet before reaching the exit point. Also the tank contents are approximately 1/10 the fission product concentration contained when the tank is ultimately filled to the design level with high activity waste. Under these conditions we see no problem with entrainment of activity into the waste tank off-gas system.

If there are any questions on the above information, please let me know.

Very truly yours,

WH-Lewis

W. H. Lewis
General Manager
Vice President

WHL/WLS/ak

Enc: DWG 8A-L-2, Rev. 8
DWG 8C-T-377, Rev. 3
DWG 15A-A-102, Rev. 4
Heat Exchanger Data Sheet
Design Calculations, 10 pages

bcc: Oliver Townsend w/o attachments
B. E. Knight w/o attachments
P. Burn w/o attachments

Knight

110733

February 15, 1967

United States Atomic Energy Commission
Division of Materials Licensing
Washington, D. C.

Attn: Mr. A. L. Wikens, Jr.
Chief, Irradiated Fuels Branch

Dear Sirs:

Drawing 8A-L-2 sent you on February 10th. on the installation of a temporary heater in the Permanent Waste Tank (SL-2), contains an error. This error, in Detail-2, shows the condensate return connection downstream of the steam trap block valve. This return line should connect between the steam trap and the downstream block valve. Attached are corrected copies of 8A-L-2, Rev. 9.

Very truly yours,

W. H. Lewis
General Manager
Vice President

WHL/[illegible]/sa

Enc: Dwg 8A-L-2, Rev. 9

bcc: Oliver Townsend w/o attachments
R. P. Wischow w/o attachments
P. Burn w/o attachments
J. J. Merkley w/o attachments
B. E. Knight w/attachments
W. H. Lewis w/attachments
M. E. Weech w/attachments

110734

RPW
MEW
EWT
BEK

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

MAR 6 1967

IN REPLY REFER TO

Docket No. 50-201

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Attention: Mr. W. H. Lewis
Vice President and General Manager

Gentlemen:

We have reviewed the information submitted with your letters of February 10 and February 15, 1967 concerning your proposed installation of a heat exchanger in Waste Tank 8D-2 and have determined that the following additional information should be submitted for our review:

1. The design modifications which will minimize the possibility of weld failures on the heat exchanger system. We discussed this point with Mr. Weech and noted that weld failures might occur in steam and condensate lines due to misalignment of the assembly. Mr. Weech informed us that design modifications to minimize this possibility would be considered.

2. The revised method for handling the condensate from the heat exchanger. As you know, the proposal to discharge condensate directly to the lagoons is not satisfactory.

Sincerely yours,

A. E. Aikens Jr.

A. E. Aikens, Jr., Chief
Irradiated Fuels Branch
Division of Materials Licensing

110735

NFS

NUCLEAR FUEL SERVICES, INC.

A SUBSIDIARY OF W. R. GRACE & CO.
BOX 124 · WEST VALLEY, N.Y. 14171
AREA CODE 716 TELEPHONE 942-3235

March 20, 1967

Mr. A.E. Aikens, Chief
Irradiated Fuels Branch
Division of Material Licensing
U.S. Atomic Energy Commission
Washington, D.C.

Dear Mr. Aikens:

In your letter of March 6, 1967, you presented two questions in relation to our proposal to boil-down the contents of Waste Tank 8D-2 using a heat exchanger installed through one of the tank nozzles. I trust that the following statements and enclosed prints will satisfactorily clarify these points.

1. Design modifications to minimize weld failures in the steam and condensate lines have been made per Mr. Weech's discussion with your staff. Six (6) copies of print 8C-T-377 Rev. 4 showing these modifications are attached.

2. The steam condensate from the heat exchanger will be collected in batches and sampled before disposal; any condensate determined to be above allowable release limits to the lagoon will be routed back to 8D-2 or the General Purpose Evaporator (7C-5). Six (6) copies of the revised drawing 8A-L-2 Rev. 10 are attached. Since each batch is sampled before discharge to the lagoon, we have effected a batch-wise disposal method for this condensate.

Should any additional questions arise, we shall be happy to supply the information required.

Very truly yours,

R.P. Wischow
Assistant General Manager

RPW:pm
Enc. Dwg. 8A-L-2 Rev. 10
8C-T-377 Rev. 4

cc: J.D. Anderson w/att.
T.C. Runion w/att.
B.G. Bechhoefer w/att.
J.J. Merkley w/att.

110736

Heat Exc[illegible]

Service [illegible] Size [illegible]
[illegible] Type [illegible] Model [illegible] Size [illegible] 8" OD
Trans. Per Shell [illegible]
Area Total [illegible]

Performance

		Tube Side
Fluid	[illegible] Salt Solution	SAT. STM. 75 PSIG
Total lb/hr	[illegible]	[illegible]
Liquid	[illegible]	—
Vapor		
Non-cond.		
Steam		[illegible]
Fluid Vap. or Cond.	[illegible]	[illegible]
Sp G Liquid	[illegible]	[illegible]
Visc Liquid CP	[illegible]	
Mol. Wt. vapors	[illegible]	[illegible]
Temp. in °F	220	320
Temp. out °F	220	320
Press in PSIG	[illegible]	75
No of Passes	Kettle	1
Vel. Ft/Sec		
Press Drop Clean PSI		
Fouling Factor Req'd.	[illegible]	[illegible]

Heat Exchanged BTU/Hr 1,855,000 Mean Temp Diff [illegible]
Transfer Rate Service 250 Clean 460 Fouling Factor [illegible]

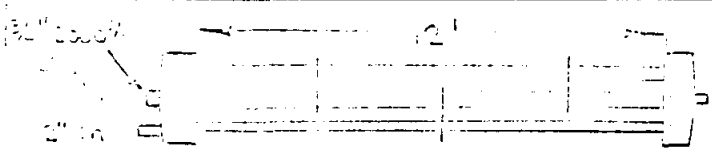
Construction

Design Press PSIG	—	150
Test Press PSIG	—	225
Design Temp °F		375

Tubes No. 30 OD in. 3/4 x 12 BWG Length ft. 12 Pitch 1" Square
Shell NONE in. id.
Connections Shell in — out —
Channel in 2" Steel Socket Weld out 1" Steel Socket Weld
Baffles 3 - 40% Cut Segmental
Code Requirements NONE ASME TEMA
wtg lbs each shell & bundle Full of water 510 Bundle 45

Part	Material	Thk. in	Corr. all	Part	Material	Thk. in	Corr.
Tubes	SA179 Steel	12 BWG	0	Tube Sheets	SA285 Steel	1/8"	0
Shell	NONE						
Shell Cover	NONE						
Channel	STEEL	1/8"	0				
Channel Cov.	STEEL	1/8"	0				
Baffles	STEEL	1/8"	0				

Remarks: 1. Tubes to be seal welded to tube sheets
2. Bundle to be used in vertical position
3. Channels and covers to be all welded and welded to [illegible]
4. Entire bundle must pass through 15 inch I.D. opening



11/[illegible] ISSUED FOR PURCHASE
11/[illegible] ISSUED FOR QUOTATION

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

IN REPLY REFER TO:
Docket No. 50-201

JAN 24 1968

Nuclear Fuel Services, Inc.
Wheaton Plaza Office Building
Suite 906
Wheaton, Maryland 20902

Change No. 6
License No. CSF-1

Attention: Mr. T. C. Runion, President

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

The telegram dated January 19, 1968, from L. D. Low to you transmitted the Atomic Energy Commission's authorization to resume processing operations at the West Valley Plant. Your letters of January 13 and 15, 1968, described the procedures to be followed at the West Valley Plant and identified certain actions to be taken and information to be supplied to comply with the requirements of the AEC letter of December 28, 1967. While it is not our intention that they remain a permanent part of Provisional Operating License CSF-1 in their present form, it is necessary to incorporate the letters of January 13 and 15, 1968, as license conditions until such time as appropriate changes to the technical specifications, License CSF-1 can be made.

In formulating these additional technical specifications required, consideration will be given to the additional information to be submitted by NFS, as described in the letters of January 13 and 15, 1968. Shortly after receipt of this information we will arrange to meet with you to discuss the additional technical specifications, and we expect at that time to be prepared to discuss other changes to the technical specifications as proposed by NFS letter of August 18, 1967.

On this basis, we have issued Change No. 6 to License CSF-1 pursuant to the Atomic Energy Act of 1954 as amended, and the regulations of Title 10, Code of Federal Regulations, Chapter 1, Parts 2 and 50 to incorporate Nuclear Fuel Services, Inc., letters of January 13 and 15, 1968, as a new Section 7.4 of the technical specifications, the text of which is attached.

Sincerely yours,

Original Signed by
J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

Attachment:
As stated above

cc: Mr. W. H. Lewis, w/attachment
Mr. B. G. Bechhoefer, w/attachment
Mr. Jon D. Anderson, w/attachment

Technical Specification 7.4
of License No. CSF-1
As Authorized by Change No. 6

7.4 In addition to the other requirements and limitations of this license, NFS shall conduct operations of the plant in accordance with the provisions of its letters to the Commission dated January 13 and 15, 1968.

110720

December 28, 1966

Mr. A. E. Aikens, Jr., Chief
Irradiated Fuels Branch
Division of Material Licensing
U. S. Atomic Energy Commission
Washington, D. C.

Dear Mr. Aikens:

Nuclear Fuel Services, Inc., West Valley, New York hereby requests a deletion be made in 5.1.2 of the Technical Specifications of CSF-1 insofar as it pertains to product loadout operations. The last four words "ON PRODUCT LOADOUT OPERATIONS" of this paragraph should be deleted.

By choosing one of two detectors, the stack particulate monitor can be sensitive to either beta or gamma emissions but neither detector is sensitive to the primary mode of decays for plutonium or uranium product. Product has been analyzed before loadout for fission activity and must meet rigid product specifications. In addition, ventilation air from the loadout hood passes through an absolute filter, two spray stages, three coarse filter stages and a final coarse and absolute filter before discharge.

We believe the requirement to have the stack monitoring system operating during product loadout operations serves no useful purpose, therefore it is felt the foregoing deletion in the Technical Specifications is justified. Should any questions arise, we shall be happy to supply additional information.

Very truly yours,

W H Lewis

W. H. Lewis
General Manager
Vice President

WHL/WJK/sk

CHANGE NUMBER - 7

WAS INCORPORATED

INTO CHANGE NO. 8

BY TELECOMMUNICATION MESSAGE

Dtd. 9-13-68, 9-17-68, & 10-17-68

SEE PAGE 2 & 3 ("VENTILATION SYSTEMS")

OF CHANGE 8

January 17, 1967

Mr. A. E. Aikens, Jr. Chief
Irradiated Fuels Branch
Division of Material Licensing
U. S. Atomic Energy Commission
Washington, D. C

Dear Mr. Aikens:

Nuclear Fuel Services, Inc. West Valley, New York hereby requests that Technical Specification 7.3 of CSF-1 be changed to read as follows:

SPECIFICATION

7.3.1 Admittance to high radiation areas where an individual might receive a dose of 100 millirem in one hour shall be controlled by locked doors with the possession of keys to these doors limited to NFS management personnel.

7.3.2 In lieu of locked doors for high radiation areas, where doors cannot readily be locked, a system such as stanchings connected by ropes with high radiation signs attached to all sides will be used. so that the radiation level at the boundary will be 5 mr/hr or less.

10CFR 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. The above procedure complies with the intent of 10 CFR 20. Some areas of the plant such as staircases that are necessary for evacuation purposes have high radiation levels from time to time. Since these areas cannot be readily locked and for safety reasons should not be locked out, we propose to use the method described in 7.3.2 to warn personnel of high radiation areas, so that they can not unintentionally enter these areas.

We believe the above change in the Technical Specifications is justified. Should any questions arise, we shall be happy to supply additional information.

Very truly yours,

W. H. Lewis
General Manager
Vice President

WHL/BEK/sk

cc: J. Anderson

bcc: M. E. Weech
W. A. Oldham
R. B. Keely

110753

NUCLEAR FUEL SERVICES, INC.
A SUBSIDIARY OF W R GRACE & CO
BOX 124 — WEST VALLEY, N.Y. 14171
AREA CODE 716 TELEPHONE 942-3235

June 3, 1963

Mr. R. B. Chitwood, Chief
Irradiated Fuels Branch
Division of Materials Licensing
United States Atomic Energy Commission
Washington 25, D. C.

Dear Mr. Chitwood:

A draft of the proposed Change No. 7 to the Technical Specifications of License CSF-1 is enclosed for your review and consideration. The specification "Zone Interface Air Velocity" is being developed and has not been included here. A brief summary of the proposed changes is included; however, I suggest that we meet at your earliest convenience to conduct a detailed review of these proposed specifications.

If you have any questions or comments, please contact me.

Very truly yours,

J. P. Duckworth, Manager
Technical Services

JPD:JRC:ah

cc: W. G. Urbon
E. D. North
B. G. Bechhoefer
D. H. Shafer

110884

December 20, 1967

Dr. John A. McBride, Director
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

The construction and installation of the tankage at Nuclear Fuel Services necessary for the storage of waste from Consolidated Edison Company's Core A fuel is now complete. We hereby request that this tankage be approved and that the restriction on processing Category III fuel be removed from Technical Specification 2.2 of Provisional License CSF-1.

This submittal provides a description and as-built drawings of the storage facility. Information and drawings were also previously submitted with:

1. Submission No. 1, Final Safety Analysis Report dated July 1, 1963

2. Submission No. 2, Final Safety Analysis Report dated June 30, 1964

3. Submission No. 21, Final Safety Analysis Report dated October 26, 1964

4. NFS letter from W. A. Rodger to A. E. Aikens dated February 17, 1965

5. NFS letter from W. H. Lewis to A. E. Aikens dated September 6, 1966.

We will be pleased to answer any further questions and to assist in any inspection of the storage facility by Division of Compliance personnel.

Very truly yours,

W. H. Lewis

W. H. Lewis
Vice President and
General Manager

WHL:JRC:ah

cc: T. C. Runion
D. H. Shafer
E. D. North

CONSOLIDATED EDISON CORE A FUEL

WASTE STORAGE FACILITY

The concentrated high level aqueous wastes, HAWC, from the processing of the Consolidated Edison's Core A fuel will be stored in an acidic waste storage tank (8D-3 or 8D-4). The waste is transferred from the high level waste neutralization tank (7D-4) via a jacketed stainless steel line to the stainless steel storage tank where it is maintained in a non-boiling condition. This stainless steel facility is located in the Waste Tank Farm area (see Drawing 40A-S-27) adjacent to the carbon steel facility (8D-1 and 8D-2).

Tankage

Each tank (see Drawing 8D-B-4 for design data) has a nominal capacity of 15,000 gallons with a maximum utilization of 90 percent. One tank will be used for waste storage while the other identical tank will be maintained as an emergency spare. Access is provided so that waste can be transferred to the spare tank in case of leakage. Another access port is provided for sampling the tanks contents.

Each tank has level, pressure and temperature indicators. A high level alarm and a high temperature alarm are also provided and are connected to the Waste Tank Farm trouble alarm which annunciates in the Control Room.

Each tank has three separate cooling coils, two vertical and one horizontal, which together are designed to remove 1,350,000 BTU/hr and maintain the waste at less than 131° F when cooling water is supplied at 80° F (see Drawing 8B-D-8). The fission product heat remaining in Con-Ed

Core A after January 1, 1968 will be less than 300,000 BTU/hr. The cooling water is normally supplied from the plant cooling water system which is connected to the emergency power system. Additional safety is provided by interconnections that permit use of the plant utility water, also connected to the emergency power system, as coolant with discharge to the storage lagoons. The cooling coils are always pressurized by either the cooling water or by utility air. A radiation detector in the cooling water return line and an air flow indicator in the utility air supply line monitor cooling coil integrity. Both instruments are connected to alarms.

Pan

The waste storage tanks sit on a stainless steel pan (see Drawing 8A-M-2) which is 18 inches high. Tank leakage would be collected in the pan which is sloped to a sump. The pan sump has a level indicator, a high level alarm and a steam jet for transferring collected solution to either 8D-3 or 8D-4. The pan sump can be sampled through an access port.

Vault

The tanks and pan sit in a reinforced concrete vault (see Drawing 8A-Q-9) which is vented to the atmosphere. A minimum of 6 feet of earth is provided above the vault for shielding.

Off-Gas System

Any acidic vapors from the waste storage tanks are passed through chemical scrubber (8C-1), knock-out drum (8D-6), high efficiency filter (8T-1 or 8T-1A) prior to discharge from the plant stack. The off-gas system for Con-Ed waste storage has been designed for 20 cfm so that

the addition of utility air will preclude the build-up of hydrogen to explosive concentrations. The piping and instrumentation of the acidic waste system is shown by Drawing 8R-A-4.

CONSOLIDATED EDISON CORE A FUEL

WASTE STORAGE FACILITY

Drawing No.	Description
8R-A-4 Rev. 5	Piping and Instrumentation Diagram
8B-D-4 Rev. 4	Con-Ed Waste Storage Tanks 8D-3 and 8D-4
8B-D-8 Rev. 1	Cooling Coil Details
8A-M-2 Rev. 2	Vault Plan and Details, Metal Liner and Inserts
8A-Q-9 Rev. 7	Vault Plan, Sections and Details
8A-L-13 Rev. 3	Piping Sections and Details
8A-L-12 Rev. 4	Piping Plan and Details
8A-L-10 Rev. 6	Piping Arrangement in Waste Tank Farm Shelter
8A-J-2 Rev. 3	Instrument Location Plan
40A-S-27 Rev. 3	Paving, Grading and Underground Lines for Waste Tankage Area

111321

NUCLEAR FUEL SERVICES, INC.
A SUBSIDIARY OF [illegible]
ROUTE 219 WEST VALLEY N.Y. 14171
AREA CODE 716 [illegible]

September 27, 1968

Mr. R. B. Chitwood, Chief
Irradiated Fuels Branch
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Chitwood:

We are hereby submitting answers to your questions concerning the stainless steel waste storage tanks to be used for the storage of waste from Consolidated Edison Company's Core A fuel. The first series of answers are the same as those provided verbally in March 1968. The second series of answers respond to your letter to W. G. Urbon on April 22, 1968. For clarity, the questions are restated before the answers. A copy of the results of the acceptance and checkout tests made on the storage tanks is also attached. I trust that this information and that submitted with our letter of December 20, 1967, is sufficient.

We hereby request that use of this tankage be approved and that the restriction on processing Category III fuel be removed from Technical Specification 2.2 of Provisional License CSF-1.

Very truly yours,

J. P. Duckworth, Manager
Technical Services

JPD:ah

Attachments

cc: E. D. North
D. H. Shafer

ANSWERS TO DIVISION OF MATERIALS LICENSING VERBAL QUESTIONS

ON NFS, WEST VALLEY, STAINLESS STEEL WASTE

STORAGE TANKS SD-3 AND SD-4

A. Construction Details

1. Q. Vault leakage--what sealing method has been used to prevent inleakage of ground water?

   A. The bottom slab has an eight mil PVC membrane. The four sides and top slab are covered with a waterproof membrane of 3 ply, 15# felt, hot-mopped asphalt ASTM-0449 Type B felt according to ASTM-D-226. Also, the four sides and top slab have a ½" thick insulating fiberboard which was hot-mopped with asphalt to protect the felt during backfilling and compaction of silty till.

2. Q. Was there a flooding test?

   A. Yes, although the construction specifications and drawings did not call for such a test.

3. Q. Is surface graded away from tank for drainage?

   A. Yes, the surface preparation is graded to slope away primarily from the Con Ed building and the ground contours are blended to prevent impounding water over the vault.

4. Q. Will the second tank float if the full tank ruptures?

   A. No, neither tank will float if either tank ruptures. This can be determined from the construction drawings. On drawing SA-Q-9 a detail of the feet are shown; these are six embedded eight inch stainless steel pipes (1'-6") one foot six inches long with anchors for hold downs. Further, the volume of one tank will not fill the vault sufficiently to float the other.

5. Q. Is the tank held down as well as up?

   A. Yes, the design and construction was such as to hold the tanks down and up. The tanks are three feet one and a half inches off the stainless steel floor and bolted down to embedded base plates.

6. Q. Is the securing mechanism corrosion resistant?

   A. Yes, the materials are stainless steel either 304 or 304L depending on the application and have been installed per tank drawings.

7. Q. Are cooling lines protected against freezing? If steam traced, what is the protection if the steam supply failed?

   A. The control building is steam heated by a wall fin tube heater and electrical auxiliary heat can be used temporarily for backup if steam system is inoperative. The main header coming from the Utility Room is buried no less than five feet underground.

8. Q. Is corrosion a problem? Are the tanks thick enough?

   A. The corrosion allowance over fifty years is 62.5 mils. According to design criteria, the tanks are thick enough.

9. Q. Have all construction methods and details been checked for stress corrosion?

   A. Yes, prior to construction, the details of stress corrosion had been carefully evaluated.

B. Quality of Materials

10. Q. Was it established that the stainless steel material was tested by Straus or Huey test to verify against carbon content and precipitation?

    A. Yes, the Huey test or electrolytic oxalic acid test was required as called for on specification C-2, Paragraph 3.2.

11. Q. Was stock material checked for freedom of defects?

    A. Yes, the standard quality control requirements of manufacturers called for such checks. The only true means would be by Ultra-Sonic Test and this would be prohibited in cost to go over all material inch by inch.

12. Q. Was it X-Rayed?

    A. No, the material was not X-Rayed except for weldments.

13. Q. Was it Ultra-Sonic tested?

    A. No, unless material manufacturer quality control required this, our requirements did not call for such a test.

C. Required Quality of Fabrication

14. Q. Has the piping and bends been tested, particularly at the bottom portion of the coil?

    A. Yes, all piping and bends have been subjected to the test requirements as outlined on specifications or drawings, particularly specification C-2.

15. Q. Have the bends been tested to check for cracking?

A. Yes, by visual inspection and by having subjected the units to the required test as called for per specification 4413-C-2.

16. Q. What Welding Code was used?

A. All welding procedures together with procedure qualification records establishing that the procedure is fully qualified were according to ASME Section IX. All welders were qualified in accordance with ASME Section IX.

17. Q. What was the qualification of the welders?

A. The welders were certified for this work and only performed work within their certification. Welder qualifications were required and reviewed per specifications.

18. Q. What welding methods were used--tungsten, inert gas?

A. The welding methods used were tungsten inert gas process and metallic arc process.

19. Q. Were any special treatments applied to minimize carbide precipitation at the weld?

A. Yes, carbide precipitation at the weld was minimized by purging with inert gas such as argon or helium and by using stainless steel wire brushes. Grinding wheels were only used on stainless steel.

20. Q. Were the welds stressed relieved?

A. No, special stress relief procedures were used.

21. Q. What temperature was used?

A. See Item 20.

22. Q. If it was annealed, was it quenched?

A. See Item 20.

23. Q. What was the upper temperature of annealing?

A. See Item 20.

D. <u>Test</u>

24. Q. What were radiography acceptance standards?

A. All butt welds on all seams and connections were 100% radiographed in accordance with the requirements of Paragraph UW51 ASME, Section VIII. Kodak type "AA" film or approved equal was used.

25. Q. For other, e.g. fittings, were dye penetrate tests made on both the root and final passes?

A. Yes, all welds not X-Rayed were dye checked according to requirements, both root and final passes.

E. In-place Testing

26. Q. Do you have an access port for coupon testing?

A. Yes, basic design provides for this and is so constructed.

27. Q. Do we have samples of the metals used for coupons testing?

A. Yes, the sample coupon method is awaiting ASMA approval.

28. Q. Do we have methods of periodically testing the spare tank (after 10 years)?

A. Yes, it is possible to do this and also to incorporate the cooling coils in the test.

F. Operability

29. Q. What tests will we run on sump transfer jet? Are there valves?

A. Water has been induced into the sump and then jetted out to the ground to simulate the actual function. As shown on the design and construction prints, there is a threeway valve.

30. Q. Are transfers feasible without exposure? Is a blank necessary?

A. Transfers can be done readily as the valving arrangement is shielded and a valve extension handle allows the operation to be semi-remote. A blank is not necessary. The threeway valve is used.

31. Q. Has the Off-Gas Systems test to determine the efficiency of scrubbers been made?

A. There does not appear to be a method of fully testing the scrubbers without filling the tank with boiling HNO_3. A standard design has been used and the quoted efficiency for these units is better than 99%.

G. Maintainability

32. Q. What is design basis on maintaining the sump jet? Can it be replaced?

A. A valve exists on both sides of the jet so that it can be unplugged with pressure should a plug occur. The jet cannot be replaced but it is feasible to install a new jet system.

33. Q. Is the valve (if present) repairable?

A. Yes, the valve can be repaired.

34. Q. How are instruments maintained?

A. Normally on a standard routine basis.

35. Q. Can the vent line be shut off to maintain the spare tank dry or under an N_2 atmosphere? Or do we have other protective methods?

A. The approved design of these tanks has a common vent system without valves so that the vent system on one tank cannot be shut off. The intent of this system is to keep a negative pressure in both tanks at all times and to maintain a constant air purge on the monitoring instrument line. By routinely monitoring the level instruments, any buildup in the tank can be detected. The purge gas on the instrument lines could possibly be changed to N_2 but we know of no reason to justify the expense of changing from the standard use of dried instrument air to N_2. The sealed vault containing the tanks is considered another protective method for extending the life of the stainless steel tanks.

36. Q. Is it possible for rain water to add carbonic acid which could unpassivate the stainless steel over a long period?

A. It is highly improbable that rain water could enter the tanks or the vault. The tanks are sealed inside a vault which would eliminate rain water from entering the tanks. The vault is a poured concrete box as described in Answers 1 and 3. The outside of the vault was given a waterproof treatment before being covered with more than six feet of silty till. As described in Answer 3, the surface was graded to further reduce the probability of rain water from entering the vault. From this, it is believed that the possibility of rain water adding carbonic acid to the outside or inside of the tank, which would unpassivate the stainless steel over a long period of time, is extremely remote.

H. Operating Criteria

37. Q. What are the criteria for waste stored in one tank to be transferred to the spare? For example: if all the cooling coils failed.

A. If all the cooling coils in the filled tank failed and the heat generating rate was sufficient to cause the liquid to exceed the maximum operating temperature of 140° F, a standby stainless steel jet would be installed in the sample opening and the solution remotely jetted from one tank to the other tank.

38. Q. If a leak rate developes of a certain size?

A. A determination would be made of the situation and the necessary emergency measures would be taken. The Safety Committee would approve all corrective actions.

39. Q. If a vault leak starts?

A. The approved actions would depend on the rate of leak and the activity content of the solution, but all efforts would be taken to stop the leak and/or repair the vault.

40. Q. At what level of buildup in sump?

A. The effect of the level in the sump on the action taken in deciding whether to transfer from one tank to another will depend on the source of leak, vault or tank. Emergency action would be started if the level exceeded 6 inches and the rate of buildup was considered excessive.

41. Q. When would you build a new vault?

A. A new vault would probably be started after a determination was made of the situation and all remedial work attempted was unsuccessful.

42. Q. What are criteria for preparedness for use of the spare tank? How long would it take to activate the spare tank?

A. It is intended to maintain the spare tank in a constant standby condition, and it is conservatively estimated that approved transfers could be initiated in less than 8 hours.

NFS

NUCLEAR FUEL SERVICES, INC.

A SUBSIDIARY OF W. R. GRACE & CO.

BOX 124 — WEST VALLEY, N.Y. 14171

AREA CODE 716 TELEPHONE 942-3235

111176

October 17, 1968

Dr. John A. McBride, Director
Division of Materials Licensing
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

Pursuant to Paragraph 50.59 of Title 10 of the Code of Federal Regulations, Nuclear Fuel Services hereby requests Change No. 8 to the Technical Specifications of Provisional Operating License CSF-1. The proposed Technical Specifications constituting change No. 8 are attached. To supplement discussions between NFS and Division of Material Licensing personnel, nuclear criticality reviews concerning 1) dissolution and feed adjustment of Core A from Consolidated Edison Company's Indian Point Reactor and 2) recycle of off-specification high enriched uranium product are also enclosed.

Nuclear Fuel Services does not believe that these changes present significant hazard considerations not described or implicit in the Final Safety Analysis submitted under Docket 50-201; therefore, authorization of the changes is requested. To permit NFS to meet its processing commitments, your earliest possible approval is desired.

Very truly yours,

W. G. Urbon
Vice President & General Manager

WGU:JRC:slk

Enclosures
Specifications: 3.0, 4.4, 4.5, 4.6, 4.8, 5.3, 6.9
2.0, 4.15, 6.3, 5.4, 6.5, 6.8 and 6.10
Nuclear Criticality Reviews

cc: E. D. North
D. H. Shafer
B. G. Bechhoefer

2.0 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).

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

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Docket No. 50-201

Change No. 8
License No. CSF-1

Nuclear Fuel Services, Inc.
Suite 906
Wheaton Plaza Building
Wheaton, Maryland 20902

Attention: Dr. E. D. North, Director
Technical Administration

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

This refers to:

(1) Nuclear Fuel Services' letter dated December 20, 1967, supplemented September 27, 1968, which requested approval to use newly installed waste tanks and authorization to process Category III fuel.

(2) Nuclear Fuel Services' submissions dated May 10, 1965, May 24, 1965, July 2, 1965, January 3, 1968, and October 17, 1968, supporting proposed changes in Technical Specifications 4.4, 4.5, and 6.10 that relate to the use of neutron absorbers in fuel dissolution and feed adjustment vessels;

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W. G. URBON

(3) The Commission's TWX dated September 13, 1968, as modified by TWX dated September 17, 1968, authorizing resumption of processing operations under interim Technical Specifications 4.14.1, 4.14.2, 4.14.3, 6.10.1, 6.10.2, 6.10.3, and 6.10.4, which is designated as Change No. 7 to Provisional License CSF-1. These specified limits to the operation of exhaust ventilation system filters and surveillance of their operation. Information submitted by NFS' letter dated October 3, 1968, provided bases for replacing these interim Technical Specifications with a new 4.14 and the revisions to Technical Specifications 5.5 and 6.5 included in NFS' submission dated October 17, 1968;

(4) Nuclear Fuel Services' request dated October 17, 1968, supplemented by TWX received October 26, 1968, for Change No. 8 to Provisional License CSF-1 involving Technical Specifications 2.0, 3.0, 4.4, 4.5, 4.8, 4.14, 4.15, 5.3, 5.4, 6.3, 6.5, 6.8, 6.9, and 6.10 in connection with the foregoing and authorization of facility modifications concerned with load-out, storage, and return for rework of highly enriched uranium product such as will be obtained from processing Category 3 fuel.

These revisions have been designated as Proposed Change No. 8, and have been considered pursuant to the provisions of Section 50.59 of 10 CFR 50.

Acid Waste Tanks

We have reviewed the installation of newly installed waste storage tanks 8D-3 and 8D-4, and have determined that with the incorporation in Provisional License CSF-1 of the Technical Specifications 5.4.1, 5.4.2, 5.4.3, 6.3.1, and 6.9.1 authorized herewith, the use of this installation to store wastes from Category 3 fuel does not present significant hazards considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by this use. Accordingly, the processing of Category 3 fuel is authorized. Specifications 2.1 and 2.2 of Provisional License CSF-1 are replaced by new Technical Specifications 3.1.1, 3.1.2, and 3.1.3.

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Since the product of the impending campaign is limited to 225 grams per liter concentration, we have for expedience revised Technical Specifications 4.8.1.3 and 4.8.1.4, which limit the concentrations in the fixed neutron absorber storage tanks, in the birdcages, and in the storage arrays, to the concentration limits authorized in the shipping approval you referenced. This is without prejudice to further revision when supported by an adequate nuclear safety evaluation.

We called to your attention the possibility that requirements for safeguarding special nuclear material product, which are being developed, may be difficult to meet in the Fuel Receiving and Storage Area. Accordingly, we have revised the technical specification authorizing birdcage storage in that area to indicate the interim character of this authorization.

We have determined that the changes to the facility pertaining to product load-out, storage, and rework of enriched uranium do not present significant hazards considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered. Operation of the facility with the changes described above is hereby authorized with the revision of Technical Specification 4.8.1 of Provisional License CSF-1 authorized herewith.

A copy of the Safety Evaluation by the Division of Materials Licensing related to the foregoing provisions of Proposed Change No. 8 to Technical Specifications is enclosed.

Editorial Revisions

The following revisions in arrangement and style of some technical specifications are included in Change No. 8. These do not involve significant safety considerations.

(1) The sequence of the "Authorized Materials" and "Definitions" specifications are reversed, to locate the definitions prior to the matters they define in the other technical specifications. The "Authorized Materials" specification has been changed to a standard format, eliminating fuel description details irrelevant to safety.

(2) The specification governing process evaporator steam pressure has been reclassified from Surveillance Requirements (6.9.1) to Safety Limits (4.15.1).

(3) The Specification 6.8.1 "Blanking-off and Locking-out" has been reworded to more accurately define the conditions subject to inspection.

Pursuant to Section 50.59 of 10 CFR Part 50, the described changes to Technical Specifications of Provisional Operating License CSF-1, Numbers 2.0, 3.1, 3.2, 3.3, 4.4, 4.5, 4.8, 4.14, 4.15, 5.3, 5.4, 6.3, 6.5, 6.8, 6.9, and 6.10 (as set forth on enclosed pages 3, 4, 5, 6, 7, 8, 20, 21, 22, 23, 24, 25, 30, 31, 32, 42a, 42b, 52, 53, 54, 55, 60, 63, 63a, 66, 67, 68, 68a, and 68b) are authorized, and all technical specifications so designated previously are rescinded.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

Enclosures:

1. Change 8 Revision, pages 3, 4, 5, 6, 7, 8, 20, 21, 22, 23, 24, 25, 30, 31, 32, 42a, 42b, 52, 53, 54, 55, 60, 63, 63a, 66, 67, 68, 68a, and 68b
2. Safety Evaluation

cc w/encls:

Mr. B. G. Bechhoefer
Mr. O. M. Ruebhausen
Mr. W. G. Urbon

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.

SAFETY EVALUATION

BY THE DIVISION OF MATERIALS LICENSING

DOCKET NO. 50-201

NUCLEAR FUEL SERVICES, INC.

AND

NEW YORK STATE ATOMIC AND SPACE DEVELOPMENT AUTHORITY

PROPOSED CHANGE NO. 8 TO TECHNICAL SPECIFICATIONS

INTRODUCTION

Nuclear Fuel Services (NFS) has requested changes in the technical specifications of License No. CSF-1 to authorize (1) the use of stainless steel storage tanks for high level liquid radioactive waste, (2) the use of fixed and soluble neutron absorbers in the dissolution and feed adjustment vessels, (3) operation with improvements in the ventilation system, including additional administrative controls, and (4) certain modifications for the load out, storage, and reworking of highly enriched uranium. These are discussed in detail below and the applicable change requests are identified in each section.

ACID WASTE STORAGE TANKS

Proposed Change

By a request dated December 20, 1967, as supplemented on September 27, 1968, NFS requested approval of the design, construction, and use of stainless steel waste storage tanks, identified as 8D-3 and 8D-4, for the storage of acidic high level radioactive liquid wastes from the processing of thoria-urania fuel from the Consolidated Edison Company's Indian Point I Core A, identified in technical specifications to the license as Category III.

Discussion

For initial operation of the plant carbon steel tanks were constructed to accommodate neutralized waste. The construction testing and emergency procedures for these tanks were set forth in the application and in our safety evaluation dated August 20, 1965, particularly in Sections 4.2, 5.4.6, 7.2. As noted therein, each tank is seated in a metal saucer within a concrete vault which would contain any leakage from the tank itself. Monitors are provided to detect leakage. In the event of leakage, the contents of the leaking tank will be transferred to a spare tank which is required for that purpose.

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To reduce faulting difficulties experienced with Dissolver Off-Gas (DOG) and Vessel Off-Gas (VOG) system filters, HEPA filter units manufactured with metal frames will be used to guard against moisture deterioration, and an additional backup HEPA filter (with spare) has been installed downstream from the exhausters in these systems, in the trunk leaking to the stack.

The operating pressure drop across the final HEPA filter in each of the ventilation exhaust system is to be limited to 85 percent of the greatest pressure drop at whic each unit has been tested.

Redundancy of filter pressure drop measurement in the Building Exhaust system will be achieved by adding instrumentation across the HEPA filter bank to the present instrumentation across the prefilter-HEPA train.

A high differential pressure alarm setting at 75 percent of HEPA filter tested pressure drop is required. A low differential alarm setting must be maintained not more than 3 inches water gage below the mean operating pressure drop across the Building Ventilation HEPA filters whenever the operating pressure drop exceeds 5 inches water gage.

No more than 10 consecutive days of operation of the plant will be permitted withou a spare Building Exhaust filter bank operable, and immediate and continued effort to repair an inoperable spare exhauster will be required.

The operability requirements for silver nitrate reactors has been changed from 0.16 to 0.6 curie of I-131 per fuel batch.

Discussion

The provision of a new backup filter installation, with the HEPA filter operating pressure drop limited to 85 percent of the tested value, is ample assurance against a major release due to filter rupture in the DOG and VOG systems. The pressure differential is monitored across the primary filters and the backup filters separately. So although flow fluctuates markedly, abnormal performance of either filter can be detected by comparison. Simultaneous rupture of the tandem filters separated by extensive ductwork and an exhauster, is unlikely.

The location of the backup filter downstream from the exhaust is not conventional. But the exhausters are located in a space vented through the Building Filtered Exhaust System and the piping to the backup filters is of welded construction and makes connections via gasketed flanges to eliminate danger of leakage where the ventilation pressure is positive.

Restricting final HEPA filters to operating pressure drops below 85% of their tested value eliminates overloading them. Prototype tests on HEPA filters indicate that they can endure impulse transients 5 times the routine test pressure drop.

Upon each installation of HEPA filters, a test is made with submicron particles to verify that the collection efficiency is greater than 99.95%. While on stream, the pressure drop across the filters is continuously monitored. Minor variations in filter resistance are not distinguishable from the pressure drop fluctuations that normally result from opening and closing confinement doors, damper adjustments, etc. An abrupt loss of a whole unit experienced in the 30 filter bank of the Building Exhaust System under relatively high filter resistance conditions lowered the pressure drop approximately 50%. Requiring the low pressure drop alarm set point to be within at least 3 inches w.g. of the mean operating pressure drop whenever the mean exceeds 5 inches w.g. should provide prompt alerting to the need for switching to the spare filter, if a major filter rupture should again occur in the Building Exhaust System. A major rupture at lower pressure drops is considered unlikely.

In addition to the redundant pressure drop instrumentation, a continuous monitor of stack effluent is provided, and is set to alarm with the accumulation of radioactive particulates equivalent to the stack release limit for one week. It alarmed on the occasion of the DOG filter rupture, but failed to register the building system filter rupture, which evidence indicates did not release a week's limit of radioactive particulates.

The connection of the monitor to the stack sampling point is via 210 feet of one-inch pipe with 16 elbows that selectively inhibit the passage of large particulates, such as could be discharged with an abrupt filter rupture. Therefore, the monitor cannot be relied upon as a filter fault detector of incidents releasing only large particles. Nor will it detect filter leakage in exhaust streams carrying little radioactivity, though it will detect abnormal concentration of small particulates in the stack gas.

We believe these provisions are adequate to warn of any major fault in the Building Exhaust filters, which are relied upon to protect the public in the event of a major dispersive accident within the plant.

The relatively low concentration of radioactive particulates impinging on the Building Exhaust filters is not sufficient to require their continuous functioning under normal circumstances, but their performance is essential to provide confinemen of upper limit accidents which have been postulated as credible within the plant.

The decrease of Building Exhaust spare filter inoperable time from 60 to 10 days is in keeping with practicable plant maintenance, and the "immediate effort" revision on spare exhauster inoperability is more conservative than the previous 60 day allowance.

With the use of a solution in the DOG chemical scrubber of greater efficiency than the caustic considered in our safety evaluation dated August 20, 1965, the increase of I-131 in a fuel batch from 0.16 to 0.6 curies before requiring the use of the silver nitrate reactor in the DOG system is unlikely to result in the reduction of the number of fuel batches which may be processed per year, under the prevailing I-131 emmission limit technical specification.

An upper limit can be established to the radioactivity available for dispersal in the event of a filter rupture. NFS favors an assumption that no more than 2% of the burden accrued on a filter would be discharged out the stack in case of filter medium rupture. Now that screens have been installed to keep filter media from entering exhausters, where it could be shredded, we feel that this is a reasonable assumption.

The activity likely to accumulate on a final HEPA filter unit is estimated by NFS not to exceed 10 curies. We agree with the method used by NFS in making this estimate, and we feel that the unshielded character of the final filter installations will impel NFS to service filter units before they accumulate this much activity.

To put this in perspective, the discharge of particulates from the stack at the technical specification limit would total about 10 curies in 3 years. Under puff release conditions 10 curies out the stack would produce a maximum ground level concentration of 5.33×10^{-4} curie seconds per cubic meter. This is equivalent to 148 hours, or less than one month, at the occupational exposure limit for strontium 90.

Applying the 2% assumed release fraction and a 7% Sr-90 composition factor, the maximum exposure at ground level would be equivalent to 1/5 hour at the occupational concentration limit.

From the foregoing evaluation we conclude that the proposed changes do not present significant hazards considerations not described or implicit in the NFS "Final Safety Analysis", and that the health and safety of the public will not be endangered.

URANIUM PRODUCT LOAD-OUT, STORAGE, AND RETURN FOR REWORK

Proposed Change

By letter dated October 17, 1968, supplemented by TWX received October 26, 1968, Nuclear Fuel Services, Inc. (NFS), requested a change in Technical Specification 4.8 of Provisional Operating License CSF-1, which establishes safety limits

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2.0 DEFINITIONS

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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.

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Material	Possession Limit	Form
Cesium-134	$6. \times 10^{-5}$ curie	any
Iodine-131	$6. \times 10^{-6}$ curie	any
Strontium-90	$6. \times 10^{-3}$ curie	any
Iodine-129	$5. \times 10^{-6}$ curie	any
Zinc-65	$2. \times 10^{-5}$ curie	any
Cobalt-60	$5. \times 10^{-2}$ curie	any
Carbon-14	$1. \times 10^{-5}$ curie	any
Tritium	$1. \times 10^{-3}$ curie	any
Any byproduct material with atomic numbers from 3 to 85 inclusive	$3. \times 10^{-5}$ curie each	any

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4.14 EXHAUST FILTER DIFFERENTIAL PRESSURE

Applicability

This specification establishes the maximum differential pressures that may be maintained across the absolute filters in the exhaust ventilation systems.

Objective

To minimize the probability of failing final filters in exhaust ventilation systems.

Specification

4.14.1 THE SUSTAINED OPERATING PRESSURE DIFFERENTIAL ACROSS ANY FINAL ABSOLUTE FILTER IN ANY OF THE EXHAUST VENTILATION SYSTEMS SHALL NOT EXCEED 85% OF THE GREATEST PRESSURE DIFFERENTIAL AT WHICH THAT FILTER HAS BEEN TESTED.

Bases

High efficiency (absolute) filters serve as the final filters in the plant's ventilation systems (Dissolver Off-Gas, Vessel Off-Gas, Main Building, Waste Tank Farm and DOG-VOG Backup). The Main Building, Waste Tank Farm and the DOG-VOG Backup absolute filters act as the final containment of radioactivity prior to discharge to the plant stack. The service of the absolute filters is limited by this Specification to minimize the probability of failing a filter. Specification 4.14.1 pertains to sustained service of the absolute filter; it is recognized that transient flow variations may cause differential pressure excursions exceeding the 85% limit as end of service life approaches.

The specifications for all absolute filters used at NFS include the common strength requirement that the absolute filter withstand a pressure drop of at least 10 inches W.G. without tearing or deforming. All absolute filters used at NFS are tested by the manufacturer to a pressure differential of at least 10 inches W.G.

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5. VENTILATION

Applicability

This specification establishes the operability requirements of the ventilation systems.

Objective

To minimize escape of airborne radioactive contaminants from process equipment and cells.

Specification

5.3.1 DISSOLVER OFF-GAS SYSTEM (DOG)--DURING DISSOLUTION OR SHEAR OPERATIONS, THE EQUIPMENT THAT SHALL BE MAINTAINED IN OPERATION INCLUDES AS A MINIMUM A CHEMICAL SCRUBBER AND ONE DISSOLVER OFF-GAS FILTER AND ITS ASSOCIATED BLOWER. IF THE FUEL BATCH CONTAINS IN EXCESS OF 0.6 CURIE OF IODINE-131, THE OPERATING EQUIPMENT SHALL ALSO INCLUDE ONE OFF-GAS HEATER FOLLOWED BY AN OFF-GAS SILVER NITRATE REACTOR. IF THIS 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)--BOTH OFF-GAS FILTERS AND ASSOCIATED BLOWERS SHALL BE OPERABLE AT THE START OF A CAMPAIGN OPERATION INVOLVING THE FEED ADJUSTMENT AND ACCOUNTABILITY TANK. THE EQUIPMENT SHALL BE OPERABLE THROUGHOUT A CAMPAIGN EXCEPT OPERATIONS MAY BE CONTINUED FOR 24 HOURS IF BOTH FILTERS AND/OR THEIR ASSOCIATED BLOWERS FAIL.

5.3.3 MAIN VENTILATION SYSTEM--BOTH SYSTEM FILTERS AND ASSOCIATED BLOWERS SHALL BE OPERABLE AT THE START OF A CAMPAIGN. EACH FILTER BANK SHALL BE MAINTAINED IN AN OPERABLE CONDITION EXCEPT FOR SERVICING PERIODS NOT EXCEEDING TEN CONSECUTIVE DAYS. SHOULD AN EXHAUSTER BECOME INOPERABLE, IMMEDIATE AND CONTINUING ATTENTION SHALL BE DIRECTED TO ITS REPLACEMENT WHILE EXERCISING MEASURES TO MAINTAIN SYSTEM CONTAINMENT. IF BOTH MAIN EXHAUST FILTER SYSTEMS BECOME INOPERABLE SIMULTANEOUSLY, IMMEDIATE AND CONTINUING ATTENTION SHALL BE DIRECTED TOWARD ORDERLY PLANT SHUTDOWN.

THE INSTRUMENTATION MONITORING THE DIFFERENTIAL PRESSURE ACROSS THE OPERATING ABSOLUTE FILTER IN THE MAIN VENTILATION SYSTEM SHALL BE OPERABLE DURING OPERATION OF THE FILTER. IF THIS INSTRUMENTATION SHOULD BECOME INOPERABLE, OPERATION OF THE FILTER MAY CONTINUE FOR 10 DAYS, PROVIDED THAT THE INSTRUMENTATION MONITORING THE DIFFERENTIAL PRESSURE ACROSS THE ABSOLUTE FILTER PLUS ROUGHING FILTER IS OPERABLE.

FILTER	PRIMARY INSTRUMENTS	ACCEPTABLE ALTERNATES
15T-49	15PDR-11	15PDR-10
	15PDAH-11	15PDAH-5
15T-49A	15PDR-6	15PDR-5
	15PDAH-6	15PDAH-4

(Change No. 8)

5.3.4 WASTE TANK OFF-GAS SYSTEM--IF A FILTER, BLOWER, OR A WASTE TANK OFF-GAS CONDENSER FAILS, THE WASTE TANK VENT SYSTEM SHALL BE SWITCHED TO THE APPROPRIATE OPERABLE STANDBY EQUIPMENT AND IMMEDIATE REPAIR SHALL BE INSTITUTED TO EXPEDITE THE RETURN OF THE FAILED EQUIPMENT TO SERVICE.

5.3.5 DOG-VOG BACKUP FILTERS--A DOG-VOG BACKUP FILTER SHALL BE OPERABLE AT ALL TIMES.

Bases

The primary control of airborne radioactivity in the plant is provided by four ventilation systems. This specification has been designed to assure the availability of sufficient ventilation capacity to provide the proper pressure differentials between operating equipment and occupied areas so that plant personnel are protected from airborne activity, and to maintain in operation essential treatment systems for the removal of radioactivity from the ventilation streams prior to their release to the atmosphere.

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-gasses pass through the scrubber, reheater, a silver nitrate reactor, a cooler, and a filter prior to blower discharge into the stack. The chemical scrubber removes about 90 percent of the iodine 131 and, if necessary, the chemical scrubber in combination with a silver nitrate reactor removes about 99.5 percent of the iodine 131. If the minimum operable equipment requirements are not met, the evolution of radioactivity is reduced by discontinuing the shearing operation and by cooling the dissolver, thereby slowing the reaction. This will reduce the fraction of the annual release limit (Technical Specification 4.1.1b) expended during equipment inoperability.

The VOG system provides equipment for the treatment of the gaseous effluents from radioactivity-containing process vessels throughout the plant. These gases are condensed, scrubbed, heated and filtered prior to blower discharge to the stack. If both VOG filter-blower units fail, operations may continue for a 24-hour period because confinement is afforded by the DOG-VOG backup filter and the Main Ventilation System, which exhausts the spaces where the vessels on the VOG system are located.

An absolute filter and a parallel alternate are installed in the common discharge line from the DOG and VOG exhausters to the stack. These filters provide in-depth protection against releases from a failed filter in either the DOG or VOG systems. Technical Specification 5.3.5 requires that one of the backup filters must be operable. Hence, the spare must be maintained in readiness.

(Change No. 8)

The Main Ventilation System controls the passage of air through the process building from clean areas to successively more contaminated areas and filters exhausted air prior to discharge up the stack. To provide for continuous operation a stand by filter-exhauster unit is provided. If one of the filter banks becomes inoperable, plant operations may continue for a 10-day period without a spare while the filter bank is being returned to service. Ten days permits filter replacement without upsetting maintenance program, yet entails minimal risk of being without spare. If an exhauster becomes inoperative, plant operations may continue while immediate and continuing attention is given to repair or replacement of the exhauster; however, the discharge through the failed exhauster must be prevented. If both exhauster-filter systems are inoperative at the same time, an orderly plant shutdown must be initiated.

Because of the potentially large inventory of radioactivity on the Main Ventilation filters, redundant instrumentation is provided. In addition to the stack monitor, performance of the operating filter is monitored by dual differential pressure recorders and high differential pressure alarms. The primary set of instruments monitors the absolute filters which are the final barrier to radioactivity release and the alternate set monitors the filter stage (absolute plus roughing).

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

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 showing that the consequences of events discussed in paragraphs 7.30 - 7.36 are acceptable within the guidelines of 10 CFR Part 100, reliance was placed on the absolute filters being effective.

(Change No. 8)

SPARE WASTE STORAGE CAPACITY

Applicability

This specification establishes the minimum provisions to be maintained to accommodate stored waste liquids, in the event of failure of storage tanks that are in use.

Objective

To prevent loss of stored radioactive waste liquids to the environment.

Specification

5.4.1 SPARE TANKS OF SUITABLE CORROSION RESISTANT MATERIAL SHALL BE MAINTAINED AT ALL TIMES WITH SUFFICIENT CAPACITY AND APPROPRIATE COOLING PROVISIONS, TO ACCOMMODATE RESPECTIVELY THE CONTENTS OF THE LARGEST TANK IN WHICH RADIOACTIVE LIQUID OF EACH CORROSIVE CHARACTERISTIC IS STORED.

5.4.2 WHEN STORAGE OF RADIOACTIVE LIQUIDS HAS BEEN INITIATED IN ANY SPARE TANK REQUIRED BY SPECIFICATION 5.4.1 OR SUCH A TANK HAS BEEN FOUND TO BE UNSUITABLE FOR SPARE SERVICE, ACTION SHALL BE INITIATED IMMEDIATELY TO REPLACE THE SPARE CAPACITY AT THE EARLIEST PRACTICAL DATE.

5.4.3 EQUIPMENT (INCLUDING PIPING, SHIELDING, LIQUID MOVER AND POWER SUPPLY) FOR TRANSFERRING THE RADIOACTIVE FLUID STORED IN ANY TANK TO THE SPARE TANKAGE RESERVED TO ACCOMMODATE IT SHALL BE MAINTAINED AT THE PLANT.

Bases

Although waste tank life expectancy is half a century, the time when a tank will fail is not accurately known. Rather than exploiting the confinement provided by the vaults in which tanks are buried during the construction of replacement tanks, this specification requires the maintenance of spare tanks, to which stored wastes could be transferred without delay.

The added margin of confinement assured by the low permeability and high ion exchange capacity of the silty till in which the tank vaults are located is sufficient to protect the public. So it is safe not to require replacement of spare tanks until they no longer serve as spares.

Since equipment for transferring the contents of the waste storage tanks to spare tankage is not built in, a technical specification requires such equipment to be available at the plant, to permit taking advantage of the provision of spare tanks.

(Change No. 8)

6.3 WASTE STORAGE TANK PAN INSTRUMENTATION

Applicability

This specification requires periodic verification of the operability of instrumentation for detecting the presence of liquid in the pans that contain the buried waste storage tanks.

Objective

To assure that liquid accumulation will be detected.

Specification

6.3.1 THE INSTRUMENTATION FOR MONITORING THE ACCUMULATION OF LIQUID IN THE PAN CONTAINING EACH WORKING LIQUID WASTE STORAGE TANK SHALL BE CHECKED FOR OPERABILITY AT LEAST ONCE PER MONTH, AND IF FOUND TO BE INOPERABLE, IMMEDIATE AND CONTINUING EFFORT SHALL BE DIRECTED TO EFFECT REPAIR.

Bases

Pans are provided in the waste tank vaults to enhance detection of tank leakage. Vault inleakage above the rim of the pan would also be accumulated in the pan. Level indicators with a high level alarm are provided to annunciate any major leakage so that remedial measures may be promptly instituted.

Since vault leakage is not expected to develop at a high rate and the silty till in which the vaults are buried is expected to inhibit migration of liquids for many years, a monthly surveillance frequency of the functioning of leak detection alarms will assure that the public will be protected from undetected tank leakage.

Failure to fulfill this specification would reduce confidence in the operability of the liquid monitoring system.

(Change No. 8)

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 ROUTINE SERVICE, WITH PARTICULATES OF AN APPROPRIATE SIZE TO ESTABLISH THAT THE INSTALLED FILTERS PROVIDE A COLLECTION EFFICIENCY OF AT LEAST 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
MAIN BUILDING	15T-49, 15T-49A
DOG-VOG BACKUP	6T-3, 6T-3A

6.5.2 THE DIFFERENTIAL PRESSURE ACROSS THE OPERATING ABSOLUTE FILTERS (OR ABSOLUTE PLUS ROUGHING WHERE THE FILTERS ARE WITHIN THE SAME FRAME) SHALL BE RECORDED BY INSTRUMENTATION.

6.5.3 ALARMS SHALL BE PROVIDED TO ANNUNCIATE HIGH DIFFERENTIAL PRESSURE ON THE OPERATING ABSOLUTE FILTERS OF THE MAIN BUILDING, WASTE TANK FARM AND DOG-VOG BACKUP SYSTEMS. THESE ALARMS SHALL BE SET TO ANNUNICATE AT LESS THAN 75% OF THE GREATEST DIFFERENTIAL PRESSURE AT WHICH THAT FILTER HAS BEEN TESTED.

6.5.4 AN ALARM SHALL BE PROVIDED TO ANNUNCIATE LOW DIFFERENTIAL PRESSURE ON THE OPERATING ABSOLUTE FILTER IN THE MAIN BUILDING VENTILATION SYSTEM. AFTER THE OPERATING DIFFERENTIAL PRESSURE EXCEEDS 5 INCHES W.G., THIS ALARM SHALL BE SET WITHIN 3 INCHES W.G. OF THE OPERATING DIFFERENTIAL PRESSURE AND SHALL BE ADJUSTED WEEKLY.

Bases

The final filters in the ventilation and off-gas systems are the primary protection against the discharge of particulate radioactivity during normal operations and particularly under accident conditions. These "absolute"

filters are certified by the manufacturer to capture at least 99.97% of all particles 0.3 microns in diameter or larger. The filters are tested after installation to detect bypass leakage as well as filter damage that may have occurred during storage and handling. Subsequent tests are not necessary as installed process instruments and the stack radioactivity monitors will indicate filter integrity and performance after the filter is placed on stream.

These tests are conducted by putting the filter-blower in service, introducing a known amount of dioctylphthalate (DOP) 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 Specification 6.5.1 is to risk the introduction of a faulty filter. In such an event, the stack monitor would quickly show up the existence of the fault and corrective action would be taken.

Specification 6.5.2 requires that the differential pressure (Δp) across the final filters in the off-gas and ventilation systems be recorded by instrumentation in the Plant Control Room or the Waste Tank Farm shelter. The differential pressure serves as an indicator of filter performance.

Alarms which annunciate high differential pressure are provided, per Specification 6.5.3, on the absolute filters which are the final containment of radioactivity. These alarms are set to annunciate at about 90% of the maximum operating differential pressure allowed by Specification 4.14. This should provide the operator with sufficient time to switch to the alternate filter; however, the differential pressure may exceed the maximum operating differential pressure limit as a transient condition during removal from service.

A low differential pressure alarm is provided for the Main Building Ventilation filters. Initially the low Δp alarm is set at about 1 inch W.G., i.e. at about 50% of the Δp experienced with a clean filter. After the Δp across the operating filter reaches 5 inches W.G., this alarm is adjusted weekly to be within 3 inches W.G. of the operating differential pressure. The purpose of this alarm is to increase the potential for detecting a failed absolute filter.

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6.9 OTHER ACTIVITY ALARMS

Applicability

This specification requires periodic verification of the operability of the alarms used to detect excessive radioactivity in cooling water or steam condensate returned from the coils or jackets used to control temperature of vessels containing radioactivity.

Objective

To provide added assurance of prompt detection of excessive radioactivity in steam condensate and cooling water.

Specification

6.9.1 THE OPERABILITY OF EACH RADIATION ALARM SYSTEM MONITORING RETURNED CONDENSATE OR COOLING WATER SHALL BE TESTED AT LEAST ONCE EVERY THIRTY DAYS BY THE APPLICATION OF A RADIATION SOURCE. IMMEDIATE AND CONTINUING EFFORT SHALL BE DIRECTED TO REPAIR ANY SYSTEM FOUND INOPERABLE.

Bases

Steam and cooling water are used for heating and cooling of process solutions. The pressures of these heat transfer fluids are higher than those of the process solution in order to minimize leakage of radioactivity if a passage should develop between a process solution and a heat transfer fluid. In addition, radiation monitors are incorporated into the piping where the steam condensate and cooling water returns to the utility room. These monitors alarm in occupied areas when excessive radioactivity is detected. Further assurance is provided by the batchwise collection and analysis of returned condensate and a radiation monitor at the outlet from the cooling coils of high level, stainless steel waste storage tanks.

Since the radiation instruments are read at least once per shift testing of the radiation alarms once per month is sufficient to verify alarm functionability.

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ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

110875

INDEXED DOCS.
50-201

DEC 28 1967

Nuclear Fuel Services, Inc.
Wheaton Plaza Building
Suite 906
Wheaton, Maryland 20902

Attention: Mr. T. C. Runion, President

Gentlemen:

On February 7, 1967, we wrote to you concerning the problems, some of them recurring, which have arisen at your company's Spent Fuel Processing Plant at West Valley, New York, since it began operations in April 1966 under AEC Provisional Operating License No. CSF-1. We recounted to you that Compliance inspections since the inception of operations had resulted in letters from the Commission to the company citing items of noncompliance with AEC rules and regulations and of other radiation safety items. We further informed you that while the company had taken corrective actions in varying degrees on the specific items brought to its attention, it was our feeling that the number and pattern of such deficiencies, if allowed to continue, could have an adverse effect on the public health and safety. We requested that you modify and improve the company's management system to establish the necessary degree of control of the plant operations to provide assurance that equipment and administrative controls were adequate.

By letters dated February 11, February 17, March 24, and November 2, 1967, you told us that corrective measures had been and would be instituted to rectify the situation. In addition, we wish to acknowledge receipt of your letters of September 18, 1967, which replied to our notice dated August 28, 1967, and of your letters dated September 27, October 19, and November 3, 1967, reporting the exposure of personnel to radioactive material.

Further inspections and investigations of your activities were conducted during the periods August 7-11, August 30-31, September 26-29, October 2-5, and October 18-20, 1967. Based on the results of these inspections and investigations, it appears that certain of your activities were not conducted in full compliance with the technical specifications of your license and the requirements of the AEC's "Standards for Protection Against Radiation," Part 20, and "Operators'

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Licenses," Part 55, Title 10, Code of Federal Regulations, as set forth in Appendix A attached to this letter. In addition to the items of apparent noncompliance found during the inspections and investigations, we wish to notify you of certain other matters relating to radiation safety which warrant your immediate attention. These matters are set forth in Appendix B.

Some of the matters noted in the appendices reflect chronic hazards to your employees. In particular, it is noted within the past year some of your employees were exposed to excessive quantities of airborne concentrations of radioactive material as a result of inadequate health physics procedures, specifically those relating to surveys and the use of protective equipment. In addition, it appears that some of the employees have not been adequately instructed in the radiation hazards involved in their jobs or that their training program has not been effective.

In view of the foregoing, specifically including the referenced appendices, it appears that radiological safety conditions in your plant are inadequate to protect the health and safety of your employees. Accordingly, we believe it is necessary that the following actions be taken by you upon receipt of this notice:

Sample stations, 1, UWA, FRS

Collin

1. Nuclear Fuel Services, Inc., (NFS) shall immediately suspend fuel processing operations in any area where personnel are required to perform any routine duty whenever smearable contamination in such area is greater than 5,000 dpm beta/100 cm^2 or is greater than 500 dpm alpha/100 cm^2, or whenever levels of radiation exist in such area that a major portion of the body could receive, in any one hour, a dose in excess of 100 mrem. For the purpose of this requirement a "routine duty" is any duty which is performed at least once in any eight-hour shift within any 24-hour period. The above contamination and radiation levels were selected because they are consistent with those specified as the upper limits for Zone III areas in your license application and referenced in the technical specifications. These requirements do not, of course, relieve NFS of the responsibility to maintain adequate control of access to all radiation areas, including those covered by paragraph 2., below, and to maintain personnel radiation exposures within 10 CFR Part 20 limits in all areas. Operations which are suspended in compliance with this paragraph may not be resumed until NFS has complied with the following:

2,000,000 50,000 α

A. The area has been decontaminated and the radiation levels reduced to the lowest practical limit. In no instance shall (1) smearable contamination be permitted which is greater

than 5,000 dpm beta/100 cm^2 or is greater than 500 dpm alpha/100 cm^2, or (2) levels of radiation exist in such area that a major portion of the body could receive, in any one hour, a dose in excess of 100 mrem.

B. The source of the contamination or radiation described in paragraph 1., above, has been determined.

C. Unless otherwise authorized by the AEC Division of Materials Licensing, appropriate equipment modifications have been completed to eliminate the source of contamination or radiation with respect to the findings as a result of B., above.

D. In order to provide opportunity for AEC inspection, Region I, Division of Compliance, has been informed by telephone 48 hours before operations are resumed in the area that the requirements of A, B, and C, above, have been completed.

2. NFS shall within 30 days perform the following with respect to personnel entries and exits to areas in which smearable contamination or radiation levels exceed those described in paragraph 1.A:

By Jan 27

A. Review and update procedures for such personnel entries and exits;

B. Review protective equipment used for such personnel entries and exits;

C. Determine whether such entries or exits result in the release of radioactive contamination to areas where personnel perform any routine duty (as defined in paragraph 1) and, if so, unless otherwise authorized by the AEC Division of Materials Licensing, complete appropriate equipment and procedural changes to prevent such contamination before further entries are made.

?

D. Institute a program to retrain personnel to upgrade their performance and discipline in the use of authorized procedures and equipment for such entries and exits; and

? MONTHLY SAFETY MEETINGS

E. Have the NFS Safety Review Committee review and approve the adequacy of actions recommended and taken in accordance with A, B, C, and D, above.

3. In addition to the provisions of paragraph 2, NFS shall within 60 days review and update all health physics procedures and equipment, have them reviewed by the NFS Safety Review Committee, and,

By Feb 26

after approval by the Committee, retrain personnel to upgrade personnel performance and discipline in the use of such procedures and equipment.

4. NFS shall promptly make an engineering analysis of the design and performance of the plant ventilation system to ascertain its adequacy to prevent the release of radioactive material to those areas in which any routine duty, as defined above, is performed by personnel. Corrective actions taken and status reports shall be included in NFS quarterly reports to the AEC.

In addition to the above, NFS shall, in accordance with 10 CFR 2.201, within thirty (30) days, file a written statement or explanation with respect to the matters set forth in Appendices A and B describing the (1) corrective steps which have been taken and the results achieved; (2) corrective steps which will be taken; and (3) the date when full compliance has been or will be achieved.

We are prepared to meet with you to discuss any matter in this letter if you believe such a meeting would be of assistance.

Very truly yours,

Original signed by:
L. D. Low

Lawrence D. Low, Director
Division of Compliance

Enclosures:
1. Appendix A
2. Appendix B

cc: Mr. W. H. Lewis
General Manager
Nuclear Fuel Services, Inc.

APPENDIX A

Inspection Conducted August 7-11, 1967

1. Contrary to 10 CFR 55.3 "Operators' Licenses," for some time prior to the inspection, two unlicensed individuals performed the functions of a senior operator. We note that since this inspection, one of the individuals has become licensed as a senior operator.

2. Contrary to license technical specification 5.1.3, stack samples were not collected and analyzed at weekly intervals, for the periods from June 6, 1967 to June 16, 1967, and from July 6, 1967 to July 14, 1967.

3. Contrary to license technical specification 7.1.4, NFS has permitted personnel to work in the laundry operation without having detailed written procedures outlining the health and safety requirements for handling contaminated items.

4. In violation of 10 CFR 20.201(b) "Surveys," since the beginning of laundry operations,

   a. adequate surveys have not been performed to evaluate exposures of laundry operators to airborne concentrations of radioactive material during the handling of contaminated items in the laundry facility, to determine compliance with 10 CFR 20.103, and

   b. no evaluation has been made of the concentrations of radioactive material released to unrestricted areas from the stacks of the laundry, to determine compliance with 10 CFR 20.106.

Investigation Conducted September 26-29, 1967, and October 2-6, 1967

1. In violation of 10 CFR 20.201(b) "Surveys,"

   a. No evaluation was made of airborne concentrations of radioactive material existing in the Extraction Sample Aisle on June 30, 1967, prior to or during the entry of a technician, to determine compliance with 10 CFR 20.103.

   b. No evaluation was made of airborne concentrations of radioactive material existing in the Extraction Sample Aisle on September 9, 1967, prior to or during the entry of a technician during a plutonium sampling operation, to determine compliance with 10 CFR 20.103.

c. An adequate evaluation was not made of the internal deposition of plutonium in a technician who had entered the Extraction Sample Aisle without respiratory protective equipment on June 30, 1967, even though nasal smears showed positive indications of possible significant plutonium ingestion, in that no survey to determine exposure was commenced until August 11, 1967.

2. Because of the failure to make an adequate survey and to assure that appropriate health physics equipment and procedures were followed, NFS, in violation of 10 CFR 20.103(a), "Exposure of individuals to concentrations of radioactive material in restricted areas," on September 9, 1967, permitted a technician to be exposed to airborne concentrations of plutonium in excess of the limits in Appendix B, Table I, of 10 CFR 20.

3. In violation of 10 CFR 20.401(b), "Records of surveys, radiation monitoring, and disposal," no record was maintained of a nasal smear survey performed on the individual on June 30, 1967, after he had exited from the area of airborne plutonium activity.

APPENDIX B

1. NFS has failed to develop an adequate means for determining the cause of contamination within the plant and to eliminate the source of such contamination to which its employees are exposed, in such areas as the Extraction Sample Aisle and the Cell Access Aisle - where high levels of contamination continued to exist for some months and employees were required to perform routine duties in the areas.

2. The overall training and orientation of employees are not adequate. As specific examples, a technician was not aware that plutonium was present in the Extraction Sample Aisle and a shift supervisor was not aware that a plutonium work restriction upon a technician previously exposed to excessive concentrations of plutonium applied to work in the Extraction Sample Aisle. Further, two employees entered the Extraction Sample Aisle on two separate occasions without wearing proper protective apparel in violation of written procedures. In addition to improving the training and orientation, management should follow-up to determine that the safety procedures are, in fact, effectively followed.

3. The program to conduct surveys in areas where the possibility of alpha contamination exists is not adequate, in that NFS possesses an insufficient number of calibrated, operable alpha survey instruments.

4. The program to conduct surveys to determine airborne concentrations of radioactive material is not adequate, in that the varying conditions which occur in the facility are not evaluated.

5. Written procedures in effect regarding entrance into and exit from the various contaminated zones are not consistent and may cause confusion for employees, specifically including Section 1.3.3 of NFS' Health and Safety Manual; a memorandum dated March 30, 1967, prepared by the Health and Safety Director; a written procedure entitled "The Effective Use of Protective Clothing;" and Job No. 52 of the Health and Safety Technician's Job Description File.

6. The bioassay program needs modification and improvements. Specifically, the procedures for issuance, follow-up, and retrieval of filled urinalysis sample bottles from employees as well as the records thereof have not been effective in some instances. An example is the loss during September 1967 of approximately 62 samples, only 38 of which were subsequently retrieved. A better

effort is necessary in obtaining more timely samples following evidence of possible internal depositions.

7. NFS should consider instituting a fecal sampling program when suspected internal depositions of plutonium have occurred. Fecal samples are of considerable value in evaluating the magnitude of a possible internal deposition and in determining whether any insoluble material is involved.

8. The method used for taking nasal smears is not consistent in that some individuals use different techniques. For example, it was noted that each employee takes his own nasal smear sample and there is no requirement that he wash his hands before taking the sample. It was noted also that some men take nasal smears before showering and some take smears after showering. These different techniques could result in the failure to detect a possible internal deposition of plutonium.

9. NFS has no established guideline for positive nasal contamination which would cause the immediate removal of an employee from risk of further exposure. Consideration should be given toward the establishment of such a policy in the more significant cases of nasal contamination, because there is a considerable waiting period between the time of submission of bioassay samples and the receipt of the results.

10. The "Lapse of Radiation Control" reports are not always prepared promptly and in some instances were completely omitted.

Lookup old limits (mag. in air supplied?)

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NFS

NUCLEAR FUEL SERVICES, INC.

A SUBSIDIARY OF W. R. GRACE & CO.
WHEATON PLAZA BUILDING, SUITE 900
WHEATON, MARYLAND 20902
AREA CODE 301-TELEPHONE 933-944

Mr. Lawrence D. Low, Director
Division of Compliance
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Low:

Pursuant to 10 CFR 2.201 and in reply to Appendix A of your letter 50-201 dated December 28, 1967, Nuclear Fuel Services, Inc. hereby submits the following statement:

General Comment

NFS believes that except possibly in the case citation 1.a. of the investigation of September 26-29, the activities described in Appendix A were in fact conducted in compliance with the requirements of the technical specifications and Part 20 and Part 55, Title 10, Code of Federal Regulations.

In the case of Citation No. 1.a., September 26-29, the possible non-compliance arose from the failure of an individual to follow NFS procedures. The individual is no longer employed by NFS.

The statement that NFS believes that in connection with practically all of the citations it was in full compliance is not intended to indicate any reluctance by NFS to remedy or improve conditions of health and safety within the facility. Our letters of January 13th and January 15th deal with our efforts to improve such conditions.

All of the citations are considered in detail below.

110870

Inspection on August 7 - 11, 1967

1. ". . . two licensed individuals performed the functions of a senior operator." NFS believes that it has been and now is in full compliance with 10 CFR 55.3, Operators Licenses. This regulation for reactors is explained and interpreted in 10 CFR 50.54 (i) through (m) part of which cannot be literally applied to a chemical processing plant. The essential feature of 10 CFR 50.54 when applied to a chemical processing plant are (l) and (m) which in effect require that a senior operator licensed pursuant to Part 55 be present at the facility or readily available on call at all times during its operation. While two shift supervisors prior to the inspection did not have senior licenses, an operator with a senior license was at all times present or readily available on call.

2. ". . . stack samples were not collected and analyzed at weekly intervals, for the periods June 6, 1967 to June 16, 1967 and July 6, 1967 to July 14, 1967." - Technical Specification 5.1.3 requires that the stack sampler be in operation at all times and that samples be collected and analyzed at least weekly. During the two periods cited, the stack monitoring system was continuously sampling the effluents and the samples were removed and analyzed each week--although not within seven day intervals. One sample was removed after eight days and the other after 10 days. There was no lapse in the surveillance of radioactive releases.

   To assure continued compliance with Technical Specification 5.1.3, a system has been installed which facilitates sample removal and the schedule for sample removal has been changed to twice per week, to insure that samples will be taken at least once each seven days.

3. ". . . NFS has permitted personnel to work in the laundry operation without having detailed written procedures outlining health and safety requirements for handling contaminated items." - Laundry operators were trained in health and safety practices including the use of appropriate radiation monitoring equipment as part of the initial orientation program. To supplement and improve this training, SOP 15-12 "Laundry Operation" was reviewed by the Plant Safety Committee on September 8, 1967, and was issued on October 13, 1967. We believe that Technical Specification 7.1.4. does not apply to plant services.

4.a. ". . .adequate surveys have not been performed to evaluate exposures of laundry operators to airborne concentrations of radioactive material during handling of contaminated items in the laundry facility." NFS believes that adequate surveys are being performed to evaluate the exposures of laundry operators to airborne radioactivity. Methods by which laundry operators are protected against exposure to excessive concentrations of airborne radioactivity include:

1) A continuous air monitor operates in the laundry. The monitor is set to alarm at 40-MPC hours and to flash an alert well below 40-MPC hours.

2) Sorting, the operation during which contamination would most probably become airborne, is done under a full exhaust hood in order to minimize airborne contamination in the breathing zone of the operator. For additional protection, the laundry operator wears a cannister mask during sorting.

3) The laundry operators are included in the plant bioassay program which includes routine urinalysis and whole-body counting.

To assure continued protection of laundry personnel, the following additional procedures are now in effect:

1) Air samples are being collected once per shift during sorting operations. A permanent air sampler which should be representative of the operator's breathing zone will be installed within the next ninety days.

2) Laundry operators have been instructed to evacuate the laundry if ventilation air stops. A sign with this instruction has been posted in the laundry. An audible alarm will be installed on the ventilation flow in the laundry to signal loss of flow.

4.b. ". . .no evaluation has been made of the concentrations of radioactive material released to unrestricted areas from the stacks of the laundry.." - An evaluation has been made of the concentrations of radioactive material released from the laundry stacks to the restricted site area. Our evaluation is continuing and we have no evidence of significant releases of radioactivity. This was to be expected since:

1) The laundry room ventilation is discharged through an absolute filter.

2) Air discharged from the laundry dryer, while running, showed airborne radioactivity statistically the same as background.

3) Air discharged from the laundry washer, while running, showed

a maximum airborne radioactivity of 3×10^{-12} μCi/ml. When this concentration, which is above the 2×10^{-12} μCi/ml control limit used at the NFS site, is averaged over a week of laundry operation, the concentration is well below the limit due to the intermittent operation of the laundry and washer.

Weekly sampling will be continued.

Investigation of September 26 - 29, 1967 and October 2 - 6, 1967

1.a. "No evaluation was made of the airborne concentrations of radioactive materials existing in the Extraction Sample Aisle on June 30, 1967 prior to or during the entry of a technician.." A Health and Safety technician entered the Extraction Sample Aisle (XSA) on June 30, 1967, in order to make a routine survey. Prior to his entry, the technician should have made an evaluation of the existing airborne radioactivity based upon the previous survey, the work being performed in related areas of the plant and the history of airborne radioactivity levels in the XSA. Notations in the Health and Safety Shift Log for June 28 - 30, 1967 indicate that the technician was aware of the existing conditions in the XSA. The technician exercised poor judgment and disregarded established Health and Safety requirements. This technician, whose actions led to citations 1a, 1c, and 3, is no longer employed at NFS.

To provide more current surveys, the schedule of air sampling the plutonium sampling stations has been increased to once per shift.

1.b. "No evaluation was made of the concentrations of radioactive material existing in the Extraction Sample Aisle on September 9, 1967 prior to or during the entry of a technician during a plutonium sampling operation . ." - An evaluation was made of airborne radioactivity in the Extraction Sample Aisle (XSA) prior to the entry of an operator and an observer on September 9, 1967. Both individuals wore respiratory protection. The NFS investigation concerning the one individual (the operator) who showed plutonium inhalation indicated that the exposure occurred due to the operator's improper disrobing. It should be noted that the operator's internal deposition was subsequently determined to be below the minimum detectable limit of whole body counting.

1.c. "An adequate evaluation was not made of the internal deposition of plutonium in a technician who had entered the Extraction Sample Aisle, without respiratory protection on June 30, 1967.." - Nuclear Fuel Services believes that a more than adequate evaluation was made of the possible internal deposition of plutonium in the Health and Safety technician who entered the Extraction Sample Aisle without respiratory protection on June 30, 1967. Urinalysis indicated that any plutonium excreta was lower than the minimum detectable concentration (0.3 dpm per liter). Whole body counting (copy of results were sent to Region I, Division of Compliance on November 10, 1967) indicated that any possible internal plutonium was below the minimum detectable content. The survey to determine exposure was initiated when positive nasal smears were detected. The failure of the Health and Safety technician, who was exposed, to fulfill his duties delayed the evaluation. However, a thorough evaluation was made. The technician is no longer employed by NFS.

2. ". . .on September 9, 1967, permitted a Technician to be exposed to airborne concentrations of plutonium in excess of the limits in Appendix B, Table I of 10 CFR 20." - The NFS investigation of the plutonium inhalation that occurred on September 9, 1967, indicated that the cause was improper disrobing and not inadequate health physics procedures. Both the operator and the observer wore respiratory protection during entry. Only the operator showed possible internal exposure. The observer had no detectable internal plutonium. The operator admitted that he had not followed the specified disrobing technique although he knew the procedure. Subsequent evaluation showed the operator's internal plutonium content to be below the minimum detectable limit of whole body counting, i. e., below 20% of a body burden.

The investigation also determined that unsatisfactory sampling equipment led to the area contamination. Significant revisions have been and are being made to the plutonium sampling equipment.

These include:

1) Use of radio-frequency sealers instead of heat sealers for closing sample bottle removal bags.

2) Use of heavy walled polyvinyl chloride bags instead of the thinner polyethylene bags for sample bottle removal.

3) Installation of sphincter seals for entry of sample bottles into the sampling glovebox.

110874

4) Movement of the permanent air sampler nearer to the work area.

5) Purchase and installation of additional zinc sulfide alpha monitors at the exits from plutonium sampling stations.

3. ". . .no record was maintained of a nasal smear survey performed on the individual on June 30, 1967, after he exited from the area of airborne plutonium activity." - The contamination of the technician on June 30, 1967 was recorded in the Health and Safety Shift Log on that date. During the investigation of the incident, the smear survey results were transcribed to a "Lapse of Control" report dated August 10, 1967 which is on file at NFS. This report was reviewed by the Division of Compliance inspectors during September 26 - 29, 1967.

Nuclear Fuel Services maintains a system of logs and records more extensive that what we believe is required in order to assure compliance with 10 CFR 20.401 (b) and the NFS license. We believe that 10 CFR 20 does not require a permanent record of nasal smears. Nasal smears, hand and foot counting and self-monitoring are items of internal control.

Very truly yours,

NUCLEAR FUEL SERVICES, INC.

T. C. Runion
President

dr

Technical Specification 7.4
of License No. CSF-1
As Authorized by Change No. 6

7.4 In addition to the other requirements and limitations of this license, NFS shall conduct operations of the plant in accordance with the provisions of its letters to the Commission dated January 13 and 15, 1968.

110760

HQ-46
(9-64)

UNITED STATES ATOMIC ENERGY COMMISSION
REGULATORY MAIL AND RECORDS SECTION

Your ☒ letter, ☐ application, dated June 3, 1968

including enclosures of Change No. 7 (proposed) to Tech. Specs.
is acknowledged and has been assigned:

DOCKET NO. 50-201 or CONTROL NO.

Please refer to the above number(s) in future correspondence.

Date Received: June 7, 1968

This is an acknowledgment form only.
It is not a reply to your communication.

CHANGE NO. 9

111111

NFS

NUCLEAR FUEL SERVICES, INC.

A SUBSIDIARY OF W. R. GRACE & CO.

BOX 124 — WEST VALLEY, N.Y. 14171

AREA CODE 716 TELEPHONE 942-3235

November 21, 1968

Dr. J. A. McBride
Director of Materials Licensing
United States Atomic Energy Commission
Washington 25, D. C.

Dear Dr. McBride:

Nuclear Fuel Services hereby requests Change No. 9 to the Technical Specifications of Provisional Operating License No. CSF-1 to permit greater flexibility in calibration of monitoring equipment. We request that Specification 3.3 be amended to include:

Material	Possession Limit	Form
Krypton-85	2 curies	gas
Krypton-85	2×10^{-1} curie	sealed source

These sources will be procured from commercial suppliers and will be used by the Health and Safety Department for calibration and functional checks of the gas section of the stack monitor. The requested radioactive quantities represent two sources in each form at concentrations appropriate for the highest exposure fuel to be processed.

We do not believe that this proposed change presents any significant hazard consideration not described or implicit in the NFS Final Safety Analysis Report and that there is reasonable assurance that the health and safety of the public will not be endangered. If further information is necessary, please contact me.

Very truly yours,

W. G. Urbon
Vice President and
General Manager

WGU:JRC:ah

cc: D. H. Shafer
E. D. North
B. G. Bechhoefer

NUCLEAR FUEL SERVICES, INC.
A SUBSIDIARY OF W. R. GRACE & CO.
BOX 124 — WEST VALLEY,
AREA CODE 716 TELEPHONE 9.

November 21, 1968

Dr. J. A. McBride
Director of Materials Licensing
United States Atomic Energy Commission
Washington 25, D. C.

Dear Dr. McBride:

Nuclear Fuel Services hereby requests Change No. 9 to the Tec Specifications of Provisional Operating License No. CSF-1 to permit g flexibility in calibration of monitoring equipment. We request that fication 3.3 be amended to include:

Material	Possession Limit	Form
Krypton-85	2 curies	gas
Krypton-85	2×10^{-1} curie	sealed source

These sources will be procured from commercial suppliers and be used by the Health and Safety Department for calibration and func checks of the gas section of the stack monitor. The requested radio quantities represent two sources in each form at concentrations appr for the highest exposure fuel to be processed.

We do not believe that this proposed change presents any sig hazard consideration not described or implicit in the NFS Final Safe Analysis Report and that there is reasonable assurance that the heal safety of the public will not be endangered. If further information necessary, please contact me.

Very truly yours,

W. G. Urbon
Vice President and
General Manager

WGU:JRC:ah

cc: D. H. Shafer
E. D. North
B. G. Bechhoefer

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

111108

Docket No. 50-201

DEC 31 1968

Change No. 9
License No. CSF-1

Nuclear Fuel Services, Inc.
Wheaton Plaza Office Building
Suite 906
Wheaton, Maryland 20902

Attention: Dr. E. D. North, Director
Technical Administration

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

This refers to your request dated November 21, 1968, for authority to receive, possess, and use krypton 85 sources to permit greater flexibility in calibration and functional checks of the gas section of the stack monitor. Your request has been designated as Proposed Change No. 9 and has been considered pursuant to the provisions of 50.59 of 10 CFR 50.

The krypton 85 to be used for calibration purposes is 0.02 percent of the amount that may be released to the environment each day under Technical Specification 4.1.1 of License No. CSF-1. On this basis we conclude that the proposed change does not present a significant hazard consideration not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change.

RECEIVED
JAN 6 1969
W. G. DIXON

111109

DEC 31 1968

Accordingly, pursuant to Section 50.59 of 10 CFR 50, the change in Technical Specification 3.3 of License No. CSF-1 as set forth on attached page No. 8 is authorized.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

Enclosure:
Change No. 9, Revision Page 8

cc: Mr. B. G. Bechhoefer, w/encl.
Mr. O. M. Ruebhausen, w/encl.
Mr. W. G. Urbon, w/encl.

3.3

Material	Possession Limit	Form
Cesium-134	$6. \times 10^{-5}$ curie	any
Iodine-131	$6. \times 10^{-6}$ curie	any
Iodine-129	$5. \times 10^{-6}$ curie	any
Strontium-90	$6. \times 10^{-3}$ curie	any
Krypton-85	3. curies	any
Zinc-65	$2. \times 10^{-5}$ curie	any
Cobalt-60	$5. \times 10^{-2}$ curie	any
Carbon-14	$1. \times 10^{-5}$ curie	any
Tritium	$1. \times 10^{-3}$ curie	any
Any byproduct material with atomic numbers from 3 to 85 inclusive	$3. \times 10^{-6}$ curie each	any

111088

NUCLEAR FUEL SERVICES, INC.
A SUBSIDIARY OF NUCLEAR SERVICES
ROCKLAND, WEST VIRGINIA 26149
AREA CODE 304 586-2000

January 22, 1969

Dr. John A. McBride, Director
Division of Materials Licensing
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

Pursuant to Paragraph 50.59 of Title 10 of the Code of Federal Regulations, Nuclear Fuel Services, Inc. hereby requests Change No. 10 to the Technical Specifications of Provisional Operating License CSF-1. Technical Specifications 3.1.1, 3.3 and 4.3 which compromise the proposed Change No. 10 and the pertinent nuclear criticality reviews are attached. We do not believe that these proposed changes present any significant hazard consideration not described or implicit in the NFS Final Safety Analysis Report submitted under Docket 50-201; therefore authorization of the changes is requested. To permit NFS to meet its processing commitments, your earliest possible approval is desired.

Very truly yours,

W. G. Urbon
Vice President and General Manager

WGU:JRC:slk

Enclosure: Technical Specification 3.1.1, 3.3 and 4.3 Nuclear Criticality Review

cc: E. D. North
D. H. Shafer
B. G. Bechhoefer

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111091

These radioisotopes, which will be purchased from commercial suppliers in liquid forms, will be in the possession and under the control of the Health and Safety Department.

C. Specification 4.3.4

It is proposed that the present Technical Specification 4.3 "Storage Canister Loading and Spacing" be modified to permit storage of fuel from the "superheater" section of the BONUS reactor. This fuel will be provided by the USAEC, Division of Production, under its contract with NFS for processing.

The fuel from the BONUS reactor consists of "boiler" and "superheater" types. Both types are Category 1 fuels. The boiler fuel satisfies the spacing requirements of Specification 4.3 but the superheater fuel assemblies (4 1/16 x 8 7/8 inches) are too large to fit within the spiders that are normally used to assure a separation of 12 inches between fuel assemblies. A criticality review (attached) has been performed which indicates an eight inch separation is more than adequate for storage of superheater assemblies. Some comments relative to the criticality review are provided below:

1. The evaluation, as noted on page 5 of the attachment, was based upon unirradiated fuel and is therefore conservative.

2. The evaluation was made relative to a $K_{\infty} = 0.90$; however, an evaluation can be made relative to $k_{eff} = 0.85$ from Figures 2 and 3. From these figures it can be seen that the k_{eff} is < 0.85 for the equivalent superheater diameter of about 7.25 inches when a separation of 8 inches of water is provided.

3. The evaluation did not consider the interaction with other stored fuel; therefore, the proposed Specification 4.3.4.2 will assure a distance of twelve inches (neutronic isolation) between superheater assemblies and assemblies of other fuel types. Practically this will be done by keeping an empty aisle on both sides of the storage section where superheater assemblies are stored.

111092

4. Mention is made in the evaluation that additional nuclear criticality considerations must be made for when the superheater assemblies are removed from beneath the water. Such considerations are already intrinsic to the processing of fuel at NFS. The canisters will be removed one at a time (therefore, one assembly at a time) and transferred to mechanical processing. The number of assemblies in the Process Mechanical Cell is limited by SOP 2-31 "PMC Handling Procedure" to

   a) not more than three in the PMC at any time.

   b) not more than one assembly at any time in the following locations: tilt fixture, north half of saw table, south half of saw table, north half Disassembly and Pushout (DIPO) table, south half of DIPO table and feed magazine.

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Material	Possession Limit	Form
Cesium-134	$6. \times 10^{-5}$ curie	any
Iodine-131	$6. \times 10^{-6}$ curie	any
Iodine-129	$5. \times 10^{-6}$ curie	any
Strontium-90	$6. \times 10^{-3}$ curie	any
Krypton-85	3. curies	any
Zinc-65	$2. \times 10^{-5}$ curie	any
Cobalt-60	$5. \times 10^{-2}$ curie	any
Carbon-14	$1. \times 10^{-5}$ curie	any
Tritium	$1. \times 10^{-3}$ curie	any
Cerium-144	$3. \times 10^{-5}$ curie	any
Ruthenium-106	$3. \times 10^{-5}$ curie	any
Zirconium-95	$3. \times 10^{-5}$ curie	any
Chromium-51	$5. \times 10^{-5}$ curie	any
Any byproduct material with atomic numbers from 3 to 85 inclusive	$3. \times 10^{-6}$ curie each	any

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Administrative procedures are established to assure correct canister loadings. The decreased water spacings permitted for Dresden canned fuel and Bonus Superheater fuel have been shown to be nuclearly safe in NFS letters to USAEC Division of Materials Licensing dated April 28, 1965 and January 22, 1968.

The consequence of exceeding this specification would be to decrease the margin of safety for the prevention of criticality. In Paragraphs 7.33 and 7.34 and 8.29 of the Safety Analysis it has been assumed that despite all design efforts, a criticality incident somehow does occur. The consequences of such an unlikely event have been analyzed and show that neither operating personnel nor the general public would be injured as a result.

The action to be taken to correct an exceeding of this specification is to take immediate steps to increase spacing of fuel in storage or decrease canister loading.

111334

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

MAR 17 1980

Nuclear Fuel Services, Inc.
Wheaton Plaza Office Building
Suite 906
Wheaton, Maryland 20902

Attention: Mr. T. C. Runion, President

Gentlemen:

Pursuant to the Atomic Energy Act of 1954, as amended, and the regulations issued pursuant thereto, it is hereby ordered that, effective immediately upon your receipt of this Order, you shall comply with the following requirements governing the transportation of licensed special nuclear materials.

These requirements apply only to those shipments of special nuclear material greater than 5,000 grams of uranium 235 (contained in uranium enriched to 20% or more in the U-235 isotope), uranium 233, or plutonium, or a combination thereof. These requirements do not apply to:

A. uranium enriched to less than 20% in the U-235 isotope;

B. special nuclear material delivering an external dose in excess of 100 rems per hour at a distance of 3 feet from any accessible surface without intervening shielding; and

C. special nuclear material that is protected pursuant to security procedures prescribed by the Atomic Energy Commission or another Government agency for classified material.

Requirement No. 1

Level I special nuclear material shall be (1) transported in the continuous personal custody of an authorized individual, or (2) delivered to a common or contract carrier for transport by such carrier in accordance with established procedures which provide for the physical protection of valuable material in transit and require an exchange of hand-to-hand receipts at origin and

destination and at all points en route where there is a transfer of custody. "Level I special nuclear material" means (1) 5,000 to 50,000 grams uranium 235 contained in uranium enriched to 20% or more in the U-235 isotope, or (2) 5,000 to 25,000 grams of uranium 233 or plutonium, or (3) 5,000 to 50,000 grams of a combination of the material specified above computed by adding the weight of U-235 to twice the total weight of U-233 and Pu.

"Authorized individual" means any individual, including an employee, a consultant, or an agent of a licensee, who has been designated in writing by a licensee to have surveillance responsibility over Level I or Level II special nuclear material or both.

Requirement No. 2

Level II special nuclear material shall be (1) transported in the continuous personal custody of an authorized individual or (2) delivered to a common or contract carrier for transport by such carrier in accordance with established procedures which provide for the physical protection of valuable material in transit and require continuous surveillance by one or more employees of the carrier of the material or the locked vehicle containing the material. "Level II special nuclear material" means any quantity of uranium 235 (contained in uranium enriched to 20% or more in the U-235 isotope), uranium 233, or plutonium exceeding the quantities specified as Level I special nuclear material.

Requirement No. 3

Records shall be kept of all shipments of Level I and Level II special nuclear material, including the means employed to protect such material while in transit.

Requirement No. 4

A report shall be made immediately to the appropriate District Safeguards Office, Division of Nuclear Materials Safeguards, by telephone, telegram, or teletype of any shipment of Level I or Level II special nuclear material that is lost or unaccounted for after the expected time of arrival.

Please refer any questions you may have to the Director, Division of Nuclear Materials Safeguards, U. S. Atomic Energy Commission, Washington, D.C. 20545.

Sincerely,

Harold L. Price
Director of Regulation

MAR 18 1969

UNITED STATES ATOMIC ENERGY COMMISSION
REGULATORY MAIL AND RECORDS SECTION

111029

☐ letter, ☒ application, dated March 17, 1969

submitting add'l information in support of your request for a change in the tecnical specifications for CSE-1 including enclosures

has been acknowledged and has been assigned:

Docket No. 50-201 or Control No. 946

Please refer to the above number(s) in future correspondence.

Date Received March 20, 1969

This is an acknowledgment form only.
It is not a reply to your communication.

NUCLEAR FUEL SERVICES, INC.
A SUBSIDIARY OF W. R. GRACE & CO.
BOX 124 · WEST VALLEY, N.Y. 14171
AREA CODE 716 TELEPHONE 942 3235

111930

March 17, 1969

Dr. John A. McBride, Director
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

The following information is submitted in support of Change No. 10 to the Technical Specifications of CSF-1, which was requested by the NFS letter dated January 22, 1969. This information is as requested in telephone conversations with personnel of Division of Materials Licensing.

A. Specification 3.1.1

Since Change No. 10 will not specify cladding material, NFS requests that "plus cladding" be deleted in the second column of the specification for Category 5 fuel. The specification would then read:

5	U - Zircaloy alloy or U - Zr alloy (U content not exceeding 10 w/o of alloy)	No limitation	Chemical

NFS also requests that the pre-irradiation U-235 enrichment in U for Category 9 fuel be changed to 2.5%. This enrichment will correspond to slightly higher than the maximum enrichment (2.1%) that NFS expects to process. The specification would then read:

9	U metal	Not exceeding 2.5%	Mechanical

B. Specification 4.3

The water retained in the structure of the Bonus superheater fuel assemblies will be removed during the first processing operation in the Process Mechanical Cell after transfer from storage. The standard operating procedure for this fuel permits only one assembly to be on the saw table at any given time; therefore, superheater assemblies which contain water will not be in close proximity in the PMC.

A nuclear criticality evaluation of storing Category 9 fuel in canisters in the storage pool is attached to this letter. The evaluation is based upon a pre-irradiation enrichment of 2.1% U-235 which is the highest

111931

enrichment of NPR fuel expected to be processed at NFS. If higher enriched fuels become available, appropriate criticality analyses will be performed. Only full canisters of outer NPR fuel tubes have been considered since this is the only available geometry for NPR fuel of the higher enrichment. Under conservative assumptions, the k_{eff} of a storage canister is 0.75 which compares favorably with the 0.85 limit imposed by Technical Specification 4.3. NFS requests that the fourth paragraph of the Bases for Specification 4.3 be changed to:

9 N - (full canister), enrichments not exceeding 1.5%

N - (*), enrichments above 1.5%

Such a change will allow NFS the flexibility of processing various NPR fuel types that the USAEC might provide yet will assure that nuclear criticality evaluations are made for canister storage of non-standard fuel assemblies.

A typographical error in the sixth paragraph of the Bases for Specification 4.3 should be corrected to "January 22, 1969."

I trust that this submission will answer your questions.

Very truly yours,

W. H. Lewis
Vice President and
General Manager

WHL:JRC:ah

Attachment

cc: D. H. Shafer

SAMPLE CALCULATION FOR k_{eff} FOR NPR FUEL STORAGE

The following core parameters were obtained from Ronald Chitwood, Richland Operations, Richland, Washington:

$\mathcal{L}^2$ = 2.182 diffusion length squared

τ = 27.367 fermi age

ε = 1.19 fast fission factor

ρ = 0.7377 resonance escape probability

Assumptions:

1. λ = 8.0 cm extrapolation length (Chitwood gave it as 7.8 but 8.0 was used as a conservative number.)
2. No credit taken for burnup of U-235
3. No credit taken for fission poisons

$\mathcal{L}_{th}$ = thermal non-leakage probability

$\mathcal{L}_f$ = fast non-leakage probability

B^2 = geometric buckling

f = thermal utilization

Fuel element size = 2.406" O.D., 1.764" I.D., 23.2" length

Canister sleeve size = 7.5" I.D., 12' length

$$B^2 = \left(\frac{2.405}{R'}\right)^2 + \left(\frac{\pi}{H'}\right)^2$$

where: $R' = R + \lambda$ $H' = H + \lambda$

$R = (3.75")(2.54\ cm/in) = 9.53\ cm$

$R' = 9.53 + 8 = 17.53$

$H = (144")(2.54\ cm/in) = 366\ cm$

$H' = 366 + 8 = 374\ cm$

111933

$$B^2 = \left(\frac{2.405}{17.53}\right)^2 + \left(\frac{3.14}{374}\right)^2$$

$$B^2 = (.137)^2 + (.0084)^2$$

$$B^2 = .0188$$

$\eta = \dfrac{\nu \Sigma_f}{\Sigma_a}$ = neutrons from fissions available to produce second generation of fissions

ν = 2.43 ("Introduction to Nuclear Engineering" by R. L. Murray, p. 105)

σ_f^{235} = 505 barns $\quad \sigma_f^{235}$ = 590 barns $\quad \sigma_a^{238}$ = 2.41 barns

Cross-sections from "Introduction to Nuclear Engineering" by R. L. Murray, p. 46.

$$\eta = \frac{(2.43)\ (.021)\ (505)}{(.021)\ (590) + (.979)\ (2.41)}$$

$$\eta = 1.74$$

$$\mathcal{L}_{th} = \frac{1}{1 + \mathcal{L}^2 B^2} = \frac{1}{1 + (2.182)\ (.0188)} = .96$$

$$\mathcal{L}_f = \exp - (B^2 \tau)$$

$$\mathcal{L}_f = e^{-(.0188)\ (27.367)} = e^{-.514} = .597$$

$$f = \frac{(\Sigma_a)_U}{(\Sigma_a)_U + (\Sigma_a)_H}$$

$$f = \frac{.239 + .0437}{.239 + .0437 + .0985}$$

$$f = .737$$

$$k_{eff} = \eta\ f\ \varepsilon\ p\ \mathcal{L}_{th}\ \mathcal{L}_f$$

$$k_{eff} = (1.74)\ (.737)\ (1.19)\ (.738)\ (.96)\ (.597)$$

$$k_{eff} = .645$$

An f of .8594 was obtained from Hanford for an optimum array of fuel elements.

Even if this f is used, k_{eff} is still only .752.

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

MAY 6 1969

Docket No. 50-201

Change No. 10
License No. CSF-1

Nuclear Fuel Services, Inc.
Box 124
West Valley, New York 14171

Attention: Mr. W. H. Lewis
Vice President and
General Manager

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. Oliver Townsend
Chairman

Gentlemen:

This refers to Nuclear Fuel Services' request dated January 22, 1969, as amended by letter dated March 17, 1969, for changes in Technical Specifications 3.1.1, 3.3, and 4.3 that would authorize: (1) receipt of higher enrichments of certain categories of fuels; (2) possession and use of additional radioisotopes for calibration sources; and (3) storage modifications to accommodate BONUS superheater fuel.

We have reviewed the proposed changes and have found that they do not present significant hazards considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change. A copy of the related Safety Evaluation by the Division of Materials Licensing is attached.

MAY 6 1969

Accordingly, pursuant to Section 50.59, 10 CFR 50, changes to Technical Specifications 3.1.1, 3.3, and 4.3 of Provisional Operating License No. CSF-1 as set forth on enclosed pages 5, 8, 17, and 18 are authorized. Page 19 is voided.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
J. A. McBride

J. A. McBride, Director
Division of Materials Licensing

Enclosures:
1. Change No. 10, Revision
   Pages 5, 8, 17, and 18
2. Safety Evaluation

cc: Mr. B. G. Bechhoefer, w/encls.
Mr. O. M. Ruebhausen, w/encls.
Dr. E. D. North, w/encls.

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Material	Possession Limit	Form
Cesium-134	$6. \times 10^{-5}$ curie	any
Cerium-144	$3. \times 10^{-5}$ curie	any
Iodine-131	$6. \times 10^{-6}$ curie	any
Iodine-129	$5. \times 10^{-6}$ curie	any
Ruthenium-106	$3. \times 10^{-5}$ curie	any
Zirconium-95	$3. \times 10^{-5}$ curie	any
Strontium-90	$6. \times 10^{-3}$ curie	any
Krypton-85	3. curies	any
Zinc-65	$2. \times 10^{-5}$ curie	any
Chromium-51	$5. \times 10^{-5}$ curie	any
Cobalt-60	$5. \times 10^{-2}$ curie	any
Carbon-14	$1. \times 10^{-5}$ curie	any
Tritium	$1. \times 10^{-3}$ curie	any
Any byproduct material with atomic numbers from 3 to 85 inclusive	$3. \times 10^{-6}$ curie each	any

4.3 STORAGE CANISTER LOADING AND SPACING

Applicability

This specification establishes limits governing fuel distribution in the Storage Pool.

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 UNIRRADIATED 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 ASSEMBLY OR PACKAGE WHICH HAS BEEN STORED UNDER WATER PRIOR TO SHIPMENT MAY BE STORED IN A CANISTER, PROVIDED THE GEOMETRY OF THE ASSEMBLY OR PACKAGE IS NOT REARRANGED.

4.3.4 CANISTERS SHALL BE SPACED TO PROVIDE A MINIMUM OF 12 INCHES OF WATER BETWEEN FUEL CONTAINED IN ADJACENT CANISTERS EXCEPT AS PROVIDED BELOW.

> 4.3.4.1 THOSE CANISTERS CONTAINING DRESDEN CANNED FUELS SHALL BE SPACED TO PROVIDE AT LEAST 9.6 INCHES OF WATER BETWEEN ADJACENT FUEL CANS AND THE IMMEDIATELY ADJACENT STORAGE RACKS SHALL REMAIN EMPTY.
>
> 4.3.4.2 THOSE CANISTERS CONTAINING BONUS SUPERHEATER FUELS SHALL BE SPACED TO PROVIDE AT LEAST 8 INCHES OF WATER BETWEEN ADJACENT CANISTERS CONTAINING SUPERHEATER FUEL AND AT LEAST 12 INCHES OF WATER BETWEEN CANISTERS CONTAINING SUPERHEATER AND THOSE CANISTERS CONTAINING OTHER TYPES OF FUEL.

Bases

The Fuel Receiving and Storage Area (FRS) has been designed to permit the handling of fuel assemblies such that geometry and administrative control prevents the interaction of one fuel assembly with another. These precautions are necessary since the FRS will at most times contain fuel in excess of that necessary to result in a criticality if placed in optimum array.

The storage racks 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. This separation of twelve inches of water prevents significant interaction of neutrons and provides a safe array.

A k_{eff} less than 0.85 provides a reasonable margin of safety to account for uncertainty in calculations and error in the identification of the fissile material content. 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 spacings permitted for Dresden canned fuel and BONUS superheater fuel have been shown to be nuclearly safe in NFS letters to USAEC Division of Materials Licensing dated April 28, 1965 and January 22, 1969.

The consequence of exceeding this specification would be to decrease the margin of safety for the prevention of criticality. In Paragraphs 7.33, 7.34 and 8.29 of the Safety Analysis it has been assumed that despite all design efforts, a criticality incident somehow does occur. The consequences of such an unlikely event have been analyzed and show that neither operating personnel nor the general public would be injured as a result.

The action to be taken to correct an exceeding of this specification is to take immediate steps to increase spacing of fuel in storage or decrease canister loading.

111340

NUCLEAR FUEL SERVICES, INC.
A SUBSIDIARY OF W. R. GRACE & CO.
BOX 124 — WEST VALLEY, N. Y. 14171
AREA CODE 716 TELEPHONE 942-3235

February 18, 1969

Dr. John A. McBride, Director
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

Pursuant to Paragraph 50.59 of Title 10 of the Code of Federal Regulations, Nuclear Fuel Services, Inc. hereby requests Change No. 11 to the Technical Specifications of Provisional Operating License CSF-1. Technical Specifications 2.0, 4.1, 4.2, 4.6, 4.7, 4.9, 4.10, 4.11, 4.12, 4.13, 4.15, 4.16, 5.1, 5.2, 5.5, 5.6, 6.1, 6.2, 6.4, 6.6, 6.7, and 7.1 are attached. We do not believe that these changes, which have been discussed in detail with your staff, present any significant hazard consideration not described or implicit in the NFS Final Safety Analysis Report submitted under Docket 50-201; therefore, authorization of the changes is requested.

If additional information is required, we will be available to meet with your staff.

Very truly yours,

W. G. Urbon
Vice President and
General Manager

WGU:JRC:ah

Enclosures: Technical Specifications

cc: E. D. North
D. H. Shafer
B. G. Bechhoefer

111041

APPENDIX A

TECHNICAL SPECIFICATION

FOR

LICENSE CSF-1

Nuclear Fuel Services, Inc.

New York State Atomic and Space Development Authority

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7.1 Administrative Requirements

INTRODUCTION

These Technical Specifications identify the significant design features, operating procedures and operating limitations which are considered important in providing reasonable assurance that the facility will be operated without undue hazard to the health and safety of either the public or plant personnel. The Technical Specifications have been grouped in seven sections whose purposes are described below.

A summary description of the processing facility is provided in Section 1.0 to aid the presentation of the Technical Specifications. Details of the facility layout, plant design, process, equipment design, methods of protecting plant personnel, methods of protecting the public and plant operation are presented in the Final Safety Analysis Report for the Nuclear Fuel Services Fuel Reprocessing Plant.

Technical terms which are commonly used at the processing plant but which may be ambiguous are defined in Section 2.0 in order to clearly indicate the intent of the various Technical Specifications.

The types and quantities of source, special nuclear and byproduct material which can be safely stored and utilized at the plant are identified in Section 3.0. These nuclear materials include irradiated fuel, unirradiated fuel for checkout or processing operations, calibration sources and laboratory standards.

The limits established in Section 4.0 define the boundaries of safe operation yet permit the flexibility essential to chemical processing. The limits have been set above the values required by normal operation but well below the values at which an accident could occur or the public safety could be jeopardized.

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 assure 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 assure the safe operation of the facility. For these reasons, if the specifications in Section 4.0, other than those for effluents 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 that would further reduce the margin of safety, an immediate shutdown is required. If the specifications for gaseous effluents or liquid effluents are exceeded, processing operations will be shutdown and immediate corrective action will be taken.

The specifications included in Section 5.0 set forth minimum conditions for safe plant operation. If specifications for monitoring gaseous and liquid effluents cannot be fulfilled, the operations which could cause a release of radioactive effluents must be shutdown, with the exception of the main ventilation system which is normally operated until the problems are corrected. Other specifications in Section 5.0 indicate primary and alternate conditions which may be fulfilled. If the alternate condition is in use the operations shall be considered to be continuing in a "ready condition." If neither primary nor alternate conditions for a particular operation can be fulfilled, the operations shall be shutdown with the exception of the waste tank off-gas system, which must be in operation while appropriate repairs are made.

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The specifications listed in Section 6.0 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 safeguard fails. The lone exception is Specification 6.1.1 requiring surveillance of the boron glass Raschig rings which are a primary safeguard in the high enriched uranium product storage tanks and the off-specification plutonium product storage tank and for which due regard is given. The other specifications in this section focus attention on controls which, while not primary safeguards, are of sufficient importance that immediate and continuing action should be made returning the failed component to service. If inspection required by Specifications 6.2 through 6.10 reveals the inoperability of any of the specified equipment, plant operation may continue under "ready condition."

Section 7.0 identifies the administrative requirements, i.e. organization standard procedures and reviews, deemed necessary for safe operation.

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2.0 DEFINITIONS

Term	Definition
Assembly	A group of elements or subassemblies combined in a structural unit. The assembly is usually that fuel structure which is removed from the reactor as an individual unit.
Area	A portion of the plant which is described by physical boundaries for identification and communication purposes.
Blanking-Off	The insertion of a removable barrier to flow in a pipe.
Campaign	The processing of a defined amount of similar nuclear fuel under a specific Letter of Authorization and Run Plan with a material inventory at the beginning and end.
Concentration Control	A technique used to assure nuclear safety that limits the concentration of fissile isotopes in process and product solutions.
Favorable Geometry	A geometry which is nuclearly safe for all credible concentrations of fissile material.
Fixed Poison Tanks	A tank or vessel equipped with neutron absorber that will not drain away, i.e. borosilicate glass Raschig rings or boron stainless steel Raschig rings.
Fissile Material	The fissile materials are uranium-233, uranium-235, plutonium-239 and plutonium-241.
Gross Count	Total alpha, beta or gamma radioactivity not classified according to specific isotope.
Locking-Out	Fixing a control switch or valve handle in either the opened or closed position by one or more padlocks or seals. The lock may be removed only by the originator or a supervisor of equal or higher authority.
MPC	The maximum permissible concentration of radioactivity in air or water to which an individual may be exposed for a specified period without exceeding regulatory limits for radiation protection.

Neutron Absorber (Poison)	A material having a high probability for capturing neutrons, e.g. boron and cadmium.
Ready Condition	A temporary condition of operation using prescribed alternate instrumentation and controls or additional administrative safeguards while immediate action is being taken to restore normal operation.
Routine Entry Area	A plant area to which entry is routinely scheduled for at least daily.
System	An integrated series of equipment and piping in which a specific function or operation is carried out.
10 CFR 20	Title 10 Part 20 of the Code of Federal Regulations which specifies the standards for protection against radiation.
U-235 Equivalent Concentration	A method of evaluating fissile material on the same basis. The formulation is not applicable to systems that depend upon geometrical control. U-235 equivalent concentration (g/l = 1.61 x Pu concentration (g/l) + 1.08 x U-233 concentration (g/l) + 1.00 x U-235 concentration (g/l).

4.1 RADIOACTIVITY CONTENT OF GASEOUS EFFLUENTS

Applicability

This specification limits the discharge of Krypton-85, Iodine-131 and radioactive particulates from the plant stack.

Objective

To provide reasonable assurance that ground level concentrations of radioactivity occurring off-site from the plant release of airborne radioactivity will meet the USAEC standards (Title 10, Part 20 Federal Code of Regulations) and the guidance values established by the Federal Radiation Council.

Specification

4.1.1 RADIOACTIVITY RELEASED FROM THE STACK SHALL NOT EXCEED:

a. KRYPTON-85	3.9×10^5	CURIES PER MONTH
b. IODINE-131	3.3	CURIES PER YEAR
c. PARTICULATE	2.7×10^2	MILLICURIES PER MONTH (gross alpha plus beta)

4.1.2 NO DISSOLUTION SHALL BE INITIATED UNLESS AN AVERAGE WIND VELOCITY OF 2 MILES PER HOUR OR GREATER SHALL HAVE PREVAILED AT THE TOP OF THE STACK OR AT THE TOP OF THE PROCESS BUILDING FOR THE PREVIOUS 30 MINUTES.

Bases

The discharge of radioactive effluents from the plant stack is discussed in paragraphs 7.6 through 7.9 of the Safety Analysis where it is shown that the significant radioactive effluents are: Krypton-85 for radiation dose from airborne radioactivity, Iodine-131 for ingestion of radioactivity via the milk chain, alpha particulates for inhalation of radioactivity and beta particulates for radiation dose from radioactivity on the ground. This analysis was based upon a design batch of fuel (20,000 megawatt-days per tonne with 150 days cooling) and average meteorological conditions (5×10^6 cubic meters per second dilution) expected for the site. The resultant limitations of Specification 4.1.1 are conservatively assigned for adequate protection of the public and may be revised based upon the results of meteorological and environmental studies.

The Krypton-85 content of a design batch of fuel is 6,300 curies. The dilution required to reduce the Krypton-85 concentration from such batches to 3×10^{-7} microcuries per cubic centimeter, the annual average maximum permissible concentration in unrestricted areas, is 4.4×10^5

cubic meters per second, i.e. less than one-tenth the average aeolian dilution expected. The limitation on Specification 4.1.1 for Krypton-85 release provides assurance that acceptable annual average concentrations will not be exceeded off-site.

The Iodine-131 content of a design batch of fuel is 1.8 curies. Off-gas treatment is provided to remove 99.5 percent of the iodine in a batch; therefore, only 9×10^{-3} curies of Iodine-131 would be released. If this amount of Iodine-131 were discharged each day (3.3 curies per year) an average dilution of 1×10^{6} cubic meters of air per second would be required to maintain pasturage conditions consistent with the guidelines of the Federal Radiation Council for milk fed to children. This dilution is about one-fifth the average aeolian dilution expected at the site. Milk produced ~~at the on-site farm~~ and at neighboring farms is routinely tested for Iodine-131 to assure that adequate dilution is provided.

The limit for particulate release is based upon the average annual aeolian dilution reducing the off-site airborne concentration to less than 2×10^{-14} microcuries per milliliter, i.e. the maximum permissible concentration specified in 10 CFR 20 for unidentified radioactivity in unrestricted areas. This limit is much more stringent than those (6×10^{-14} for Plutonium-239 and 2×10^{-12} for unidentified beta emitters) corresponding to the most radiotoxic materials actually present in the effluents from the processing plant.

The wind velocities recorded in the plant's Control Room will be used to determine compliance with Specification 4.1.2. The wind velocity at the top of the stack will be the primary instrument; the wind velocity at the process building roof will be used if the stack instrument or its recorder are inoperative. An acceptable wind velocity will be deemed to have existed in the last thirty minutes if during this period the area of the wind velocity trace above 2 mph is greater than the area of the wind velocity trace less than 2 mph.

4.2 RADIOACTIVITY CONTENT OF LIQUID EFFLUENTS

Applicability

This specification establishes the upper limit to the radioactivity that may be contained in liquid effluents.

Objective

To limit the radioactivity in Cattaraugus Creek at any time to a safe concentration.

Specification

4.2.1 BATCHES OF LIQUID WASTE DETERMINED BY MEASUREMENT TO CONTAIN LESS THAN 3×10^{-7} MICROCURIES PER MILLILITER GROSS ALPHA PLUS GROSS BETA RADIOACTIVITY MAY BE DISCHARGED TO THE ENVIRONMENT WITHOUT ROUTING THROUGH THE STORAGE LAGOONS.

4.2.2 NO MORE THAN 96 CURIES OF UNIDENTIFIED ALPHA PLUS BETA ACTIVITY OTHER THAN TRITIUM MAY BE TRANSFERRED TO THE STORAGE LAGOONS DURING ANY CALENDAR QUARTER. TRANSFERS MUST BE IN SEPARATE BATCHES WHICH MAY NOT CONTAIN MORE THAN 6 CURIES OF UNIDENTIFIED ALPHA PLUS BETA ACTIVITY AS DETERMINED BY MEASUREMENT.

4.2.3 LIQUID WASTE RELEASED TO THE CATTARAUGUS CREEK FROM ALL SOURCES ON THE SITE SHALL BE REGULATED SO THAT (A) THE CONCENTRATION OF RADIOACTIVITY AT THE MONITORING STATION IN CATTARAUGUS CREEK SHALL NOT AT ANY TIME EXCEED TWICE THE CONCENTRATION SPECIFIED IN COLUMN 2, TABLE II, APPENDIX B OF 10 CFR 20 AND (B) THE ANNUAL TIME-AVERAGE CONCENTRATION OF RADIOACTIVITY IN CATTARAUGUS CREEK SHALL NOT EXCEED THAT SPECIFIED IN COLUMN 2, TABLE II, APPENDIX B OF 10 CFR 20.

Bases

Title 10, Part 20 of the Code of Federal Regulations (10 CFR 20) specifies the maximum permissible concentration (MPC) of radioactive effluents to unrestricted areas. The MPC for Strontium-90, one of the most radiotoxic isotopes released in the effluent from an irradiated fuel processing plant, is 3×10^{-7} microcuries per milliliter above natural background. The batch limit of Specification 4.2.1 assures safe releases by conservatively assuming all the radioactivity is Strontium-90 and by ignoring the natural radioactivity content.

Specification 4.2.2 provides in-depth control to the release of radioactive effluents. Radioactive plant liquid discharges are collected batchwise in a plant interceptor where the liquid is sampled and analyzed. Activity content above the 6 curie limit is either returned to the plant

for further processing or directed to a high level waste storage tank. Activity contents below the six curie limit may be transferred to storage lagoons, which consist of three separate lagoons in series with a total capacity of about 5,000,000 gallons, prior to controlled release to Cattaraugus Creek. The ninety-six curies allowed each quarter for transfers of radioactivity to the storage lagoons is less than the annual discharge allowed to the Cattaraugus Creek even if all the radioactivity was Strontium-90 and is only about twenty percent of allowable annual discharge corresponding to the creek flows and the isotopic compositions of discharges experienced during 1967 - 1968.

Column 2, Table II, Appendix B of 10 CFR 20 specifies the maximum permissible concentration of radioactivity in liquid effluents to unrestricted areas; these specified concentrations are annual time-averaged concentrations. Specification 4.2.3 recognizes these limits as appropriate for the NFS facility but imposes additional safeguards by requiring the concentration at any time be less than twice the 10 CFR 20 limits. NFS intends to provide additional safety by discharging well below the limit.

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4.12 SOLID RADIOACTIVE WASTE BURIAL

Applicability

This specification establishes the limits to be observed in the transfer and storage of solid radioactive waste material resulting from the separation, in the reprocessing plant, of nuclear material from nuclear fuel. This specification also applies to the burial of contaminated plant equipment if buried in the same area as separation waste.

Objective

To assure retention of activity from buried waste.

Specification

4.12.1 RADIOACTIVE ~~SOLID PROCESS~~ WASTE GENERATED BY THE OPERATION OF THE PLANT SHALL BE BURIED COMPLETELY WITHIN SILTY TILL. BEFORE A NEW EXCAVATION 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 WASTE 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 WASTE SHALL BE FOUR FEET EXTENDING BELOW THE TOP OF THE UNDISTURBED SILTY TILL STRATUM. SUITABLE MEANS SHALL BE PROVIDED AND MAINTAINED TO MINIMIZE EROSION IN THE DEFILES, BETWEEN THESE DEFILES AND BURIAL AREAS AND OF THE EARTH COVER AFTER THE EXCAVATIONS HAVE BEEN FINALLY BACKFILLED.

4.12.2 THE LOCATION AT WHICH RADIOACTIVE SOLID WASTE ARE BURIED IN ACCORDANCE WITH SPECIFICATION 4.12.1 SHALL BE MARKED WITH CONCRETE CAIRNS. A PLOT SHOWING THE APPROXIMATE LOCATION OF ALL WASTE BURIED SHALL BE MAINTAINED ACCOMPANIED BY AN INDEX DESCRIBING THE GENERAL TYPES OF WASTE 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.

4.12.3 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

Nuclear Fuel Services operates two waste burial areas at the West Valley site. One area is for plant generated waste and is licensed under the regulations of the U.S. Atomic Energy Commission. The other area is

primarily for waste generated at facilities other than the processing plant and is licensed under the regulations of the State of New York. This specification applies to the burial area subject to USAEC licensing and regulations.

As pointed out in Paragraph 2.45 of Safety Analysis, the location specified for the burial of separation waste is of such a favorable geological formation that any radioactivity 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 Safety Analysis, the ion exchange and permeability data for the soil in which the radioactive waste is to be buried indicate that the radioactivity 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 isotope, 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 soil to an adjacent water course. This specification requires that at least four feet of silty till cover the waste.

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

The waste transport vehicles are surveyed both before and after use to prevent contamination and minimize radiation exposure of transport personnel.

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.

4.13 CAUSTIC CONCENTRATION IN CARBON STEEL WASTE STORAGE TANKS

Applicability

This specification establishes limits for the minimum concentration of caustic in the carbon steel waste storage tanks.

Objective

To assure that the liquid waste in the carbon steel waste storage tanks shall be maintained in alkaline condition at all times.

Specification

4.13.1 EXCESS CAUSTIC (BASED ON THE STOICHIOMETRIC AMOUNT REQUIRED TO NEUTRALIZE ACIDIC WASTE) IN THE CARBON STEEL HIGH LEVEL WASTE STORAGE TANKS SHALL BE PRESENT IN CONCENTRATION OF (A) AT LEAST 10% BASED ON THE WASTE VOLUME INTRODUCED THEREIN UNTIL THE TOTAL VOLUME OF INTRODUCED WASTE EXCEEDS 10,000 GALLONS, (B) AT LEAST 5% ON THE SAME BASIS UNTIL THE TOTAL VOLUME OF INTRODUCED WASTE EXCEEDS 100,000 GALLONS AND (C) AT LEAST 1% ON THE SAME BASIS AFTER THE VOLUME OF INTRODUCED WASTE EXCEEDS 100,000 GALLONS.

Bases

To minimize the corrosion rate of the carbon steel high level liquid waste storage tanks, the normally acidic liquid wastes are neutralized prior to transfer into the tanks. In addition, excess caustic is maintained in the tanks to prevent corrosion resulting from any inadvertent addition of acidic waste. The procedure for maintaining excess caustic is as follows: the waste storage tanks will be initially filled one-third to one-half 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 percent excess caustic will be maintained in the tank for waste solution additions above 100,000 gallons.

The amount of caustic could be reduced considerably still leaving an excellent margin of safety. 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, the general public would still be protected.

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4.16 RESPIRATORY PROTECTION EQUIPMENT

Applicability

This specification establishes the requirements necessary to protect plant personnel from airborne concentrations of radioactive material exceeding the maximum permissible concentrations of the 10 CFR 20 for restricted areas.

Objective

To assure that plant personnel will not inhale excessive quantities of radioactive material.

Specification

4.16.1 FULL FACE RESPIRATORS APPROVED FOR RADIOACTIVE MATERIALS UNDER BUREAU OF MINES, SCHEDULE 21B, SHALL BE USED WHERE THE CONCENTRATION OF AIRBORNE RADIOACTIVITY ANTICIPATED DURING OCCUPANCY MAY EXCEED 40 MPC-HOURS BUT WILL NOT EXCEED 800 MPC-HOURS.

4.16.2 SELF-CONTAINED BREATHING APPARATUS SATISFYING THE BUREAU OF MINES SCHEDULE 13 REQUIREMENTS OR SUPPLIED AIR RESPIRATORS SATISFYING THE BUREAU OF MINES SCHEDULE 19 SHALL BE USED IN ATMOSPHERES WHERE THE CONCENTRATION ANTICIPATED DURING OCCUPANCY MAY EXCEED 800 MPC-HOURS.

4.16.3 PRIOR TO EACH ENTRY INTO A CONTAMINATED ATMOSPHERE, INDIVIDUALS WEARING RESPIRATORY PROTECTION SHALL CHECK THE MASKS FOR FIT AND LEAKAGE.

4.16.4 FOLLOWING EACH USE, RESPIRATORY PROTECTION MASKS SHALL BE RETURNED FOR DECONTAMINATION UNDER APPROVED HEALTH AND SAFETY PROCEDURES. WHEN THE CLEANING AND REPAIR HAS BEEN APPROVED BY HEALTH AND SAFETY PERSONNEL, MASKS SHALL BE PACKAGED INDIVIDUALLY IN PLASTIC BAGS AND DELIVERED FOR REUSE WITH CLEAN CLOTHING SUPPLIES.

4.16.4.1 MASKS SHALL NOT BE RELEASED FOR REUSE IF FIXED RADIOACTIVE CONTAMINATION EXCEEDS 100 CPM BETA/PROBE AREA OR 100 CPM ALPHA/PROBE AREA.

4.16.4.2 FILTER CANISTERS FOR MASKS SHALL NOT BE RELEASED IF (A) SMEARABLE CONTAMINATION EXCEEDS 50 DPM ALPHA/100 CM^2 OR 500 DPM BETA/100 CM^2 OR (B) FIXED RADIOACTIVE CONTAMINATION EXCEEDS 500 CPM BETA/PROBE AREA AT ONE INCH.

Bases

The 40 MPC-Hours concentration is the concentration of airborne radioactivity that a worker could breath throughout his work week and not inhale excessive radioactivity. NFS requires that if plant personnel

may be exposed to such a concentration, no matter how short the exposure time, respiratory protection must be worn. NFS limits the use of filter masks to twenty times the 40-hour MPC concentration even though these high efficiency filters provide a protection factor of 100.

For use in airborne concentrations exceeding 800 MPC-hours, NFS requires the use of supplied air equipment which is approved by the Bureau of Mines, a recognized authority in respiratory protection.

The contamination limits of reuse of masks and mask canisters are consistent with the limitations for uncontaminated plant areas (Zone II) and are expressed in radiation units used at the plant. Specification 4.16.4.1 is based upon (1) a 20% counting efficiency and 50 cm^2 probe area for beta monitoring and (2) a 50% counting efficiency and 75 cm^2 probe area for alpha monitoring. Specification 4.16.4.2 is based upon (1) a mask area of 178 cm^2 and (2) a minimum of 2,500 cpm per 1 mrad/hr.

5.1 [illegible]

Applicability

This specification identifies the equipment, frequency of sampling and the analyses required for monitoring radioactive gaseous and liquid plant effluents.

Objective

To appraise radioactive liquid and gaseous effluents.

Specification

5.1.1 WHENEVER LIQUID IS DISCHARGED FROM THE STORAGE LAGOONS, A REPRESENTATIVE SAMPLE SHALL BE CONTINUOUSLY WITHDRAWN BY THE CATTARAUGUS CREEK SAMPLER. AT LEAST EVERY SEVEN DAYS, A COMPOSITE OF THIS SAMPLE SHALL BE ANALYZED FOR TRITIUM, GROSS ALPHA AND GROSS BETA RADIOACTIVITY.

IF THE CONTINUOUS SAMPLER BECOMES INOPERABLE, UP TO 2,000,000 GALLONS MAY BE DISCHARGED FROM THE LAGOONS IN ANY ONE YEAR PERIOD PROVIDED CATTARAUGUS CREEK SAMPLES ARE COLLECTED NEAR THE SAMPLER LOCATION AND ANALYZED DURING EACH LAGOON DISCHARGE OR FOR EACH 100,000 GALLONS OF DISCHARGE, WHICHEVER RESULTS IN THE GREATER SAMPLING FREQUENCY.

ANALYSES SHALL BE SUFFICIENT SENSITIVITY TO APPRAISE CONFORMANCE WITH SPECIFICATION 4.2.3 AND MAY INCLUDE ASSAY OF SPECIFIC NUCLIDES.

5.1.2 THE STACK SAMPLER LOCATED BESIDE THE STACK WHICH COLLECTS A REPRESENTATIVE SAMPLE OF THE STACK GAS SHALL BE IN OPERATION AT ALL TIMES AND THE SAMPLES COLLECTED SHALL BE ANALYZED AT LEAST EVERY SEVEN DAYS TO VERIFY CONFORMANCE WITH THE RELEASE LIMITS FOR I-131 AND PARTICULATE RADIOACTIVITY. IF THE STACK SAMPLER SHOULD BECOME INOPERATIVE, IMMEDIATE REPAIR SHALL BE INSTITUTED TO EXPEDITE THE RETURN OF THE STACK SAMPLER TO SERVICE AND THE STACK MONITOR SHALL BE USED IN THE INTERIM TO DETERMINE PARTICULATE AND I-131 RELEASES.

THE STACK GAS SHALL BE CONTINUOUSLY MONITORED FOR I-131, KR-85 AND PARTICULATE RADIOACTIVITY. IF THE CONTINUOUS MONITOR FAILS, A SAMPLE COLLECTED THROUGHOUT EACH SHIFT SHALL BE ANALYZED FOR I-131 AND PARTICULATE RADIOACTIVITY; KR-85 MAY BE CALCULATED FROM FUEL BURNUP DATA.

Bases

About one-half mile downstream of confluence of Cattaraugus Creek and Buttermilk Creek, a sampling station withdraws a sample of the creek water. The samples are collected and a portion is analyzed at least weekly for gross alpha, gross beta and tritium activity. Each month

a composite is made from the weekly samples and analyzed by a commercial laboratory for gross beta, gross alpha and Strontium-90 radioactivity. These radioactive concentrations are compared to their respective maximum permissible concentrations to show conformance with Specification 4.2. The sampler like any such equipment is subject to mechanical failure; therefore, a limited amount of intermittent sampling is permitted in place of the continuous sampler.

Iodine-129 is present in such low concentrations in plant effluents that after dilution in Cattaraugus Creek it cannot be detected by presently available analytical methods; therefore, the Iodine-129 concentration in the Cattaraugus Creek is calculated from the analyses of Lagoon 3 composite sample and the dilution of the lagoon discharges by Cattaraugus Creek.

Gaseous plant effluents are sampled in the plant stack. For the stack sampler, air is drawn through a particulate filter and an activated charcoal filter to remove particulate and Iodine-131 radioactivity. The samples are removed from their location alongside the plant stack at least every seven days and are analyzed to show conformance with Specification 4.1.

For the stack monitor, air is drawn from the stack to the Upper Extraction Aisle when radiation monitors continuously determine the particulate, Iodine-131 and Krypton-85 radioactivity. The stack monitor signals an alarm in the control room when excessive gaseous radioactivity is released.

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5.5 HIGH RADIATION AREA ACCESS

Applicability

This specification authorizes alternatives that may be used in lieu of the control device specified in 10 CFR 20.203 (c) (2).

Objective

To assure that personnel are warned before they enter areas where the radiation exposure potential may be significant.

Specification

5.5.1 LOCKED DOORS SHALL BE USED, WHERE FEASIBLE, TO CONTROL ADMITTANCE TO HIGH RADIATION AREAS WHERE AN INDIVIDUAL MIGHT RECEIVE A DOSE IN EXCESS OF 100 MILLIREM IN ONE HOUR. POSSESSION OF KEYS TO THESE LOCKS SHALL BE LIMITED TO NFS SUPERVISORY PERSONNEL.

5.5.2 IN LIEU OF LOCKED DOORS OR THE CONTROL DEVICE SUCH AS SPECIFIED IN 10 CFR 20.203 (c) (2), A SYSTEM OF STANCHIONS CONNECTED BY ROPES WITH "HIGH RADIATION AREA" SIGNS ATTACHED SHALL BE USED TO OBSTRUCT ALL ACCESSIBLE SIDES OF A HIGH RADIATION AREA.

Bases

10 CFR 20.203 requires that a means be provided to warn personnel when they enter areas where the radiation exposure potential may be significant. The above procedure complies with the intent of 10 CFR 20. Some areas of the plant such as staircases that are necessary for evacuation purposes have high radiation levels periodically. Since these areas cannot be readily locked and for safety reasons should not be locked out, the method described in this specification is used to warn personnel of high radiation areas so that they cannot unintentionally enter these areas.

5.6 CONTAMINATION

Applicability

This specification establishes the smearable contamination limits for routine entry areas.

Objective

To maintain adequate radiological conditions for the protection of the health and safety of plant personnel.

Specification

5.6.1 IRRESPECTIVE OF THE USE OF PROTECTIVE CLOTHING, MAINTENANCE OF CONTAMINATED ZONE BOUNDARIES, AND THE APPLICATION OF EXITING TECHNIQUES IN ACCORDANCE WITH PLANT PROCEDURES, ANY AVERAGE SMEARABLE CONTAMINATION IN A ROUTINE ENTRY AREA EXCEEDING 50,000 DPM BETA/100 CM^2 OR 500 DPM ALPHA/100 CM^2 MUST BE REDUCED BELOW THESE LEVELS WITHIN SEVEN DAYS SINCE DETECTION AND THE SOURCE OF THE CONTAMINATION MUST BE TERMINATED. IF THIS REDUCTION IS NOT ACHIEVED, (A) PROCESSING OPERATIONS REQUIRING ACCESS TO THE CONTAMINATED AREA SHALL BE DISCONTINUED UNTIL THE CONTAMINATION SOURCE IS TERMINATED AND THE CONTAMINATION IS REDUCED BELOW THE ABOVE LEVELS AND (B) THE USAEC DIVISION OF COMPLIANCE SHALL BE NOTIFIED WITHIN 48 HOURS.

5.6.2 THE FOLLOWING AREAS ARE EXEMPT FROM 5.6.1: FRS CASK HANDLING AREA, SCRAP REMOVAL ROOM, HOT SHOP, HULL BURIAL, OFF-GAS BLOWER ROOM AND EXTRACTION CHEMICAL ROOM (RECOVERED ACID AREA).

Bases

All work areas that personnel enter daily are smeared at the step-off pads at least once per shift by Operations personnel and are surveyed by Health and Safety personnel at least weekly. Upon detecting significant contamination, the area is surveyed by Health and Safety to determine the extent and the levels of contamination. The limits of the contamination area are clearly marked and roped off. Operations personnel begin cleanup within at least twelve hours and complete cleanup within 7 days. If the cleanup cannot be completed within 7 days, processing operations in the area are suspended until the area is clean and the USAEC, Region I, Division of Compliance is notified.

The 500 dpm alpha/100 cm^2 and 50,000 dpm beta/100 cm^2 limits reflect the smearable contamination levels which could cause the airborne radioactivity to exceed the maximum permissible concentrations for 40 hours per week exposure. The limits are conservatively based upon Plutonium-239 and Strontium-90, respectively.

Certain plant areas are exempt from the specified limits. The respiratory protection appropriate for the contamination level is worn and special procedures are used for work in these areas.

This program recognizes that periodically contamination will occur due to human or mechanical failure yet puts due emphasis on eliminating reoccurring contamination.

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6.4 [illegible]

Applicability

This specification establishes the surveillance requirements for the standby utility equipment.

Objective

To assure that standby utility equipment 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 MONTHS TO DETERMINE THE AUTOMATIC START CAPABILITY AND THE 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. THE NEXT CAMPAIGN SHALL NOT START UNTIL THE EQUIPMENT IS RETURNED TO SERVICE.

6.4.2 THE EQUIPMENT LISTED BELOW SHALL BE OPERATED AT LEAST ONCE EVERY THREE MONTHS TO DETERMINE PERFORMANCE FOR AT LEAST ONE HOUR UNDER AN OPERATING LOAD. ANY DEFICIENCIES WILL BE CORRECTED AS SPECIFIED IN 6.4.1.

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

6.4.3 THE QUANTITY OF DIESEL FUEL IN THE OIL STORAGE TANK (31D-2) SHALL, DURING NORMAL OPERATIONS, BE MAINTAINED AT 8,000 GALLONS OR MORE OF DIESEL FUEL. IF USED DURING ABNORMAL CONDITIONS, OPERATIONS MAY CONTINUE AS LONG AS A VOLUME OF 4,000 GALLONS OR GREATER IS MAINTAINED. A VOLUME LESS THAN THIS LATTER VALUE WILL REQUIRE THE COMMENCEMENT OF PLANT SHUTDOWN WHICH SHALL CONTINUE UNLESS THE PRIMARY FUEL (NATURAL GAS) SERVICE IS RESTORED OR THE FUEL OIL SUPPLY IS REPLENISHED.

Bases

Standby equipment has been provided to enhance the reliability of utilities supplied to safety equipment and devices. In order to insure satisfactory performance, the standby equipment is 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 at least 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 standby services.

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Export Controlled Information

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Export Controlled Information

7.1 ADMINISTRATIVE REQUIREMENTS

Applicability

This specification establishes administrative standards for the operation of the plant.

Objective

To provide an administrative system embodying managerial principles of demonstrated effectiveness in achieving safety of operation.

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 INCLUDING, BUT NOT LIMITED TO, THE FOLLOWING:

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

7.1.1.2 AN OPERATIONS MANAGER, REPORTING TO THE GENERAL MANAGER OR TO AN ASSISTANT GENERAL MANAGER WITHOUT RESPONSIBILITY FOR HEALTH AND SAFETY OR TECHNICAL SERVICES, 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 OR TO AN ASSISTANT GENERAL MANAGER WITHOUT RESPONSIBILITY FOR OPERATIONS, RESPONSIBLE FOR MONITORING THE RADIOLOGICAL SAFETY OF ALL PLANT ACTIVITIES AND FOR ADVISING ALL DEPARTMENTS ON RADIOLOGICAL SAFETY MATTERS.

7.1.1.4 A TECHNICAL SERVICES MANAGER, REPORTING TO THE GENERAL MANAGER OR TO AN ASSISTANT GENERAL MANAGER WITHOUT RESPONSIBILITY FOR OPERATIONS, RESPONSIBLE FOR REVIEW OF PROPOSED CHANGES IN DESIGN WITH RESPECT TO RADIOLOGICAL AND NUCLEAR CRITICALITY SAFETY AND RESPONSIBLE FOR A CONTINUAL REVIEW OF OPERATIONS TO ASSURE NUCLEAR CRITICALITY SAFETY.

7.1.1.5 A PLANT SAFETY COMMITTEE, REPORTING TO THE GENERAL MANAGER, RESPONSIBLE FOR BOTH RADIOLOGICAL NUCLEAR CRITICALITY SAFETY BY (A) REVIEW AND APPROVAL OF ALL OPERATING PROCEDURES, LETTERS OF AUTHORIZATION AND EMERGENCY PROCEDURES, (B) REVIEW AND APPROVAL OF ALL CHANGES IN THE PROCESS (OTHER THAN THOSE PERMITTED IN THE LETTER OF AUTHORIZATION OR OPERATING PROCEDURES) AND EQUIPMENT DESIGN AND (C) INVESTIGATION OF ABNORMAL OCCURRENCES AND RECOM-

MINIMIZATION OF MAGNITUDE WHICH WILL PRECLUDE THEIR EXPLOITATION. THE COMMITTEE SHALL CONSIST OF, BUT NOT TO BE LIMITED TO, THE TECHNICAL SERVICES MANAGER, THE HEALTH AND SAFETY DIRECTOR AND THE OPERATIONS MANAGER OR THEIR DESIGNATED ALTERNATES. ALL THREE MEMBERS OR THEIR ALTERNATES MUST BE PRESENT IN ORDER TO CONDUCT OFFICIAL BUSINESS.

7.1.2 WRITTEN PROCEDURES SHALL BE MAINTAINED UNDER THE OVERALL DIRECTION OF THE GENERAL MANAGER COVERING THE:

7.1.2.1 PREPARATION, APPROVAL AND 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 BE LIMITED TO, SHUTDOWN OF SPECIFIC EQUIPMENT, ACTIVATION OF THE EMERGENCY PLAN, SUMMONING OF OUTSIDE SUPPORT AND RE-ENTRY TO THE ACCIDENT AREAS.

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

7.1.3 PLANT OPERATION. MEMBERS OF THE PRODUCTION OPERATING STAFF SHALL HAVE A THOROUGH TRAINING IN THE PROCEDURES GOVERNING THE WORK FOR WHICH THEY ARE RESPONSIBLE. DRILLS SHALL BE HELD AT LEAST YEARLY. A CURRENT SET OF OPERATING PROCEDURES SHALL BE PROVIDED IN 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, WRITTEN PROCEDURES FOR HANDLING ABNORMAL OPERATING CONDITIONS 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 INCLUDE MINUTES OF MEETINGS OF THE PLANT SAFETY COMMITTEE AS WELL AS DETAILS OF INVESTIGATIONS AND RECOMMENDATIONS IN REGARD TO ABNORMAL OCCURRENCES AND ACTIONS TAKEN THEREON.

7.1.6 AUDIT OF OPERATIONS. MANAGEMENT SHALL PROVIDE FOR ANNUAL AUDITS OF THE OPERATIONS THROUGHOUT THE ENTIRE PLANT WITH REGARD TO RADIOLOGICAL AND NUCLEAR SAFETY. THESE AUDITS SHALL BE PERFORMED BY COMPETENT PERSONNEL NOT DIRECTLY RESPONSIBLE FOR THE OPERATION AUDITED.

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

7.1.7.1 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. THE OPERATION MAY CONTINUE IN A "READY CONDITION" TEMPORARILY UNLESS THERE IS ANOTHER OCCURRENCE THAT WOULD FURTHER REDUCE THE MARGIN OF SAFETY IN WHICH EVENT IMMEDIATE SHUTDOWN IS REQUIRED.

7.1.7.2 IMMEDIATE STEPS SHALL BE TAKEN TO ESTABLISH THE CAUSE OF THE OCCURRENCE IN 7.1.7.1 BY RESPONSIBLE MANAGEMENT REVIEW.

7.1.7.3 IF SPECIFICATIONS 4.1.1 OR 4.2.3 ARE EXCEEDED, (A) THE PLANT SYSTEM CAUSING THE EXCESSIVE RADIOACTIVITY RELEASE SHALL BE SHUTDOWN, (B) REGION I DIVISION OF COMPLIANCE SHALL BE NOTIFIED AND (C) THE OPERATION OF THE SYSTEM SHALL NOT BE RESUMED UNTIL THE SOURCE OF THE RELEASE HAS BEEN DETERMINED AND THE NECESSARY CORRECTIVE ACTION HAS BEEN TAKEN. IF THE RELEASE IS FOUND TO EXCEED THE AMOUNT ALLOWED UNDER THE SPECIFICATION FOR A ONE YEAR PERIOD, OPERATION SHALL NOT BE RESUMED WITHOUT SPECIFIC AUTHORIZATION OF THE USAEC.

7.1.8 EVALUATIONS MADE OF PROVISIONS FOR NUCLEAR CRITICALITY SAFETY SHALL UTILIZE THE DOUBLE CONTINGENCY CRITERIA TO ESTABLISH THAT APPROVED NUCLEAR CRITICALITY SAFETY SYSTEMS AND/OR PROCEDURES CANNOT RESULT IN NUCLEAR CRITICALITY UNLESS TWO OR MORE INDEPENDENT AND UNLIKELY MISHAPS OCCUR.

NFS

NUCLEAR FUEL SERVICES, INC.

BOX 124 · WEST VALLEY, N.Y. 14171
AREA CODE 716 TELEPHONE 942-3235

September 17, 1969

Dr. John A. McBride, Director
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. McBride:

Pursuant to Paragraph 50.59 of Title 10 of the Code of Federal Regulations, Nuclear Fuel Services, Inc. hereby requests Change No. 11 to the Technical Specifications of Provisional Operating License CSF-1. The proposed changes contained herein are substantially those proposed by NFS in its submittal dated February 18, 1969; however, modifications have been made to previously proposed changes to incorporate comments and directions of the Irradiated Fuels Branch. Technical Specifications 2.0, 4.1, 4.2, 4.6, 4.7, 4.9, 4.10, 4.11, 4.12, 4.13, 4.15, 4.16, 5.1, 5.2, 5.5, 5.6, 6.1, 6.2, 6.4, 6.6, 6.7, and 7.1 are attached.

NFS believes that these changes, which have been discussed in detail with personnel of the Irradiated Fuels Branch, do not contain any significant hazard consideration not described or implicit in the Final Safety Analysis Report submitted under Docket 50-201; therefore, authorization of the changes is requested.

If additional information is necessary, we will be available to meet with your staff.

Very truly yours,

J. P. Duckworth

J. P. Duckworth
Plant Manager

JPD:JRC:js

Enclosures: Technical Specifications

cc: E. D. North
D. H. Shafer
B. G. Bechhoefer

ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JUN 19 1970

Docket No. 50-201

Change No. 11
License No. CSF-1

Nuclear Fuel Services, Inc.
Wheaton Plaza Office Building
Suite 906
Wheaton, Maryland 20902

Attention: Dr. E. D. North, Director
Technical Administration

AND

New York State Atomic and Space
Development Authority
230 Park Avenue
New York, New York 10017

Attention: Mr. James Cline
General Manager

Gentlemen:

This refers to your request dated June 16, 1970, for authority to receive, possess, and use up to five millicuries of strontium 85 as a tracer in investigation and evaluation of liquid waste treatment systems. Your request has been designated as Proposed Change No. 11 and has been considered pursuant to the provision of 50.59 of 10 CFR Part 50.

We have concluded that the proposed change does not present a significant hazard consideration not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the change.

Accordingly, pursuant to Section 50.59 of 10 CFR 50, the change in Technical Specification 3.3 of License No. CSF-1 as set forth on attached page No. 8 is authorized.

FOR THE ATOMIC ENERGY COMMISSION

Lyall Johnson
Acting Director
Division of Materials Licensing

Enclosure:
Change No. 11, Revision Page 8

cc: Mr. B. G. Bechhoefer, w/encl.
Mr. O. M. Ruebhausen, w/encl.
Mr. J. P. Duckworth, w/encl.

Material	Possession Limit	Form
Cesium-134	$6. \times 10^{-5}$ curie	any
Cerium-144	$3. \times 10^{-5}$ curie	any
Iodine-131	$6. \times 10^{-6}$ curie	any
Iodine-129	$5. \times 10^{-6}$ curie	any
Ruthenium-106	$3. \times 10^{-5}$ curie	any
Zirconium-95	$3. \times 10^{-5}$ curie	any
Strontium-90	$6. \times 10^{-3}$ curie	any
Strontium-85	$5. \times 10^{-3}$ curie	any
Krypton-85	3. curies	any
Zinc-65	$2. \times 10^{-5}$ curie	any
Chromium-51	$5. \times 10^{-5}$ curie	any
Cobalt-60	$5. \times 10^{-2}$ curie	any
Carbon-14	$1. \times 10^{-5}$ curie	any
Tritium	$1. \times 10^{-3}$ curie	any
Any byproduct material with atomic numbers from 3 to 85 inclusive	$3. \times 10^{-6}$ curie each	any

SAFETY EVALUATION
BY
DIVISION OF MATERIALS LICENSING
NUCLEAR FUEL SERVICES, INC.
DOCKET NO. 50-201
TRACER USE OF STRONTIUM 85

By TWX dated June 16, 1970, Nuclear Fuel Services, Inc. requested a change in Technical Specification 3.3 of License CSF-1, to authorize 5 millicuries of strontium-85 for use as a tracer in investigation and evaluation of liquid waste treatment systems.

Technical Specification 3.3 of Provisional Facility Operating License No. CSF-1 lists nuclides, quality limits, and forms which are authorized for receipt, possession, storage, and use for standards, tests, measurements, and calibrations.

By telecon June 18, 1970, I verified that the proposed tracer investigations would be laboratory work and not environmental dispersal studies. Hence, any waste disposal from these investigations would be subject to the prevailing technical specifications in the license which govern these matters. The reason for the request to authorize use of strontium-85 instead of the already authorized strontium-90 is to permit prompt assessment of the presence of strontium in the fluids and solids used in the experimental investigation. Strontium-90 assays require several weeks each to obtain data read out.

The concentration tabulated in Column 2, Table II, Appendix B, 10 CFR Part 20 for soluble strontium-85 is 300 times the concentration for soluble strontium-90. If all 5 millicuries of the 65-day half life strontium-85 should be disposed of via liquid effluent system, the impact on the operation of the plant liquid waste discharge system under the prevailing technical specifications would be imperceptible. Each day's throughput of waste water would dilute 5 millicuries of strontium-85 to less than 1/3 the concentration listed in Column 2, Table II, Appendix B, 10 CFR 20.

Therefore, I conclude that authorizing the use of the requested strontium-85 for the purpose indicated does not present a significant hazard consideration not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by authorizing the requested change.

W. H. Ray
W. H. Ray
Irradiated Fuels Branch
Division of Materials Licensing

DATE: JUN 19 1970

Wheaton
September 4, 1970

TO: E. A. Wick

FROM: E. D. North

SUBJECT: DML Meeting of September 4, 1970

A meeting was held with DML on September 4, 1970, to discuss the preliminary license information recently submitted for the Plutonium Development Facility. The meeting was attended by:

DML	NFS
Dr. Mann	Dr. North
Mr. Johnson	Mr. Wick
Mr. Chitwood	Mr. Brennan
Mr. Nussbaumer	
Mr. Rouse	
Mr. Lipinski	

The following subjects were discussed:

Natural Phenomena

DML has not yet received an evaluation of the Dames and Moore Report on earthquakes and this subject was deferred. A single set of earthquake criteria will be used for the West Valley site.

Dr. Mann stated that the tornado criteria were satisfactory except for the pressure drop of 2 psi in 3 seconds. DML is fixed on 3 psi in 3 seconds. NFS requested justification for 3 psi and received none.

There was considerable discussion of what constitutes a Class I structure. DML feels that the employees within the Class I structure must be considered part of the public. Multiple barriers must be present between plutonium and employees. NFS' position is that this will be achieved for nitrate solutions but that it is not possible to have multiple barriers in the glove box line. Solution container design must assure that major spills will not result from the occurrence of natural phenomena. This does not mean a full blown Class I dynamic analysis, however.

Natural Phenomena (Continued)

DML would like additional description of the flood problem. We should look at the Corp of Engineers method for determining maximum credible flood; DML will supply the method. This does not appear to be a significant problem.

NFS should justify the position that plutonium materials in the Class II structure are truly non-dispensible. The possibility of fires and explosions must be considered.

Three areas that should be given special attention are: (1) pellet sintering furnace; (2) tanks, vessels or lines carrying nitrate solutions; (3) scrap recovery area. An accident analysis should be made for these areas.

There was much discussion on the filter question. NFS stated that the final filter on the exit stream would be Class I. Mann would like two Class I final filters since a single barrier is not fool proof. This will require careful study. Several options are available: (1) install a backup final filter; (2) mount glove box exit filters to resist earthquakes; (3) duct filters ahead of final filters that are shock resistant.

There was some discussion of the procedure for buttoning up the facility when a tornado is imminent. Mann was not impressed by the buttoning-up technique and requested an evaluation of the consequences of a tornado strike without warning. If the consequences are reasonable, this approach will probably be acceptable although we should be prepared to defend the concept. An earthquake almost certainly will occur without warning, however, the consequences are probably acceptable.

Nussbaumer requested information on the instrumentation on the hydrogen supply to the sintering furnace to permit safe shutdown following an earthquake or tornado.

NFS stated that the criticality alarm system would be functional through the earthquake or tornado. A quality assurance program will be required for the Class I system as a minimum. See Appendix B, 10 CFR 50. NFS does not believe this program should be extended beyond Class I structures. Some description of the standard NFS procedures should be provided for sensitive areas such as: (1) poisoned tanks; (2) vessles and lines for Pu nitrate solutions, etc.

The storage area for completed fuel assemblies must be capable of withstanding earthquake forces without rearrangement that could develop a critical mass.

NFS will take the position that absolute filters will withstand 3 psi under all conditions.

E. D. North

EDN/kac

cc: Messrs. R. W. Deuster
J. L. Brennan
W. A. Oldham
W. H. Lewis
J. R. Clark

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

Oct 16, 1970

Docket No. 50-201 Change No. 12

Nuclear Fuel Services, Inc.
ATTN: Dr. E. D. North, Director
Technical Administration
Suite 906
Wheaton Plaza Office Building
Wheaton, Maryland 20902

AND

New York Atomic and Space Development
Authority
ATTN: Mr. James Cline
General Manager
230 Park Avenue
New York, New York 10017

Gentlemen:

This refers to Nuclear Fuel Services' letter dated March 1970, supplemented July 1970, September 4, 1970, September 21, 1970 and October 8, 1970, supporting proposed changes in Technical Specifications 4.14, 5.3, 6.4 and 6.5 that relate to operation of the new head-end ventilation system. These revisions have been designated by NFS as Proposed Change No. 13 and have been considered pursuant to the provisions of Section 50.59 of 10 CFR 50.

We have reviewed the operation of the new head-end ventilation system. As we informed you, the following additional information, based on operating experience, must be provided in a timely manner:

1. A 6- and 12-month report giving information on filter loading and personnel exposure for operation of this new facility.

2. A 6-month report summarizing your 1500-gallon sump tank monitoring program experience.

We have determined that the changes to the facility pertaining to the new head-end ventilation system, as submitted, do not present significant

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hazards considerations not described or implicit in the NFS "Final Safety Analysis Report" and there is reasonable assurance that the health and safety of the public will not be endangered.

A copy of the Safety Evaluation by the Division of Materials Licensing related to the foregoing provisions of your Proposed Change No. 13 and revised as Change No. 12 is enclosed.

Pursuant to Section 50.59 of 10 CFR Part 50, changes to Technical Specifications of Provisional Operating License CSF-1, Numbers 4.14, 5.3, 6.4 and 6.5 (as set forth on enclosed pages 42a, 52, 53, 54, 54a, 54b, 54c, 61, 62, designated previously are rescinded.

FOR THE ATOMIC ENERGY COMMISSION

(Original Signed by
Richard E. Cunningham)
for Lyall Johnson
Acting Director
Division of Materials Licensing

Enclosures:
1. Chg. No. 12, Rev. Pgs. 42a, 52, 53, 54, 54a, 54ab, 54c, 61, 62, 63, 63a and 63b

2. Safety Analysis

cc: Mr. B. G. Bechhoefer, w/encl.
Mr. O. M. Ruebhausen, w/encl.
Mr. J. P. Duckworth w/encl.

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4.14 EXHAUST FILTER DIFFERENTIAL PRESSURE

Applicability

This specification applies to the maximum differential pressures that may be maintained across the absolute filters in the exhaust ventilation systems.

Objective

To minimize the probability of filter failure in exhaust ventilation systems.

Specification

4.14.1 THE SUSTAINED OPERATING PRESSURE DIFFERENTIAL ACROSS ANY ABSOLUTE FILTER IN ANY OF THE EXHAUST VENTILATION SYSTEMS SHALL NOT EXCEED 85% OF THE GREATEST PRESSURE DIFFERENTIAL AT WHICH THAT FILTER HAS BEEN TESTED.

Basis

High Efficiency (absolute) filters are used in the plant's ventilation systems (Dissolver Off-Gas, Vessel Off-Gas, Main Building, Head End, Waste Tank Farm, and DOG-VOG Backup). The absolute filters act as the final containment of radioactivity prior to discharge to the plant stack. The service of the absolute filters is limited by this specification to minimize the probability of filter failure. Specification 4.14.1 pertains to sustained service of the absolute filter; it is recognized that transient flow variations may cause differential pressure excursions exceeding the 85% limits as end of service life approaches.

The specifications for all absolute filters used at NFS include the common strength requirement that the absolute filter withstand a pressure drop of at least 10 inches W. G. without tearing or deforming. All absolute filters used at NFS are tested by the manufacturer to a pressure differential of at least 10 inches W. G.

5.3 VENTILATION

Applicability

This specification applies to the operability requirements of the ventilation systems.

Objective

To assure that ventilation equipment necessary for safety is operable during plant operation.

Specification

5.3.1 DISSOLVER OFF-GAS SYSTEM (DOG)—DURING DISSOLUTION OR SHEAR OPERATIONS, THE EQUIPMENT THAT SHALL BE MAINTAINED IN OPERATION INCLUDES AS A MINIMUM A CHEMICAL SCRUBBER AND ONE DISSOLVER OFF-GAS FILTER AND ITS ASSOCIATED BLOWER. IF THE FUEL BATCH CONTAINS IN EXCESS OF 0.6 CURIE OF IODINE-131, THE OPERATING EQUIPMENT SHALL ALSO INCLUDE ONE OFF-GAS HEATER FOLLOWED BY AN OFF-GAS SILVER NITRATE REACTOR. IF THIS 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)—BOTH OFF-GAS FILTERS AND ASSOCIATED BLOWERS SHALL BE OPERABLE AT THE START OF A CAMPAIGN OPERATION INVOLVING THE FEED ADJUSTMENT AND ACCOUNTABILITY TANK. THE EQUIPMENT SHALL BE OPERABLE THROUGHOUT A CAMPAIGN EXCEPT OPERATIONS MAY BE CONTINUED FOR 24 HOURS IF BOTH FILTERS AND/OR THEIR ASSOCIATED BLOWERS FAIL.

5.3.3 AT THE START OF A PROCESSING CAMPAIGN IN THE PROCESS MECHANICAL CELL, THE FOLLOWING EQUIPMENT SHALL BE OPERATIVE:

HEAD-END VENTILATION SYSTEM

1. BOTH ELECTRICALLY-DRIVEN EXHAUST BLOWERS.
2. THE SPARE PROPANE GAS ENGINE-DRIVEN EXHAUST BLOWER.
3. ALL FILTER STAGES IN BOTH FILTER BANKS.

MAIN PLANT VENTILATION SYSTEM

1. ONE EXHAUST BLOWER AND ITS ASSOCIATED ROUGHING AND ABSOLUTE FILTERS.

5.3.4 AT THE START OF A PROCESSING CAMPAIGN IN THE SOLVENT EXTRACTION AREA, THE FOLLOWING EQUIPMENT SHALL BE OPERATIVE:

MAIN PLANT VENTILATION SYSTEM

1. BOTH EXHAUST BLOWERS AND THEIR ASSOCIATED ROUGHING AND ABSOLUTE FILTERS

HEAD END VENTILATION SYSTEM

1. ALL FILTER STAGES IN ONE FILTER BANK.

2. EITHER BOTH ELECTRICALLY-DRIVEN EXHAUST BLOWERS OR THE SPARE PROPANE GAS ENGINE-DRIVEN EXHAUST BLOWER.

5.3.5 IF, FOR ANY REASON, THE REDUNDANCY IN FILTRATION CAPABILITY OR EXHAUST BLOWER AVAILABILITY OF EITHER THE HEAD-END OR MAIN PLANT VENTILATION SYSTEM IS LOST FOR A PERIOD OF TIME EXCEEDING 10 DAYS, ALL PLANT OPERATIONS EXCEPT THE AVAILABLE VENTILATION SYSTEM SHALL BE SHUT DOWN IN AN ORDERLY FASHION. PLANT OPERATIONS MAY BE CONTINUED BEYOND THE 10 DAY PERIOD ONLY IF THE FAILED EQUIPMENT INVOLVES A MAJOR COMPONENT SUCH AS A BLOWER OR A DRIVE UNIT PROVIDED (1) THE SAFETY COMMITTEE HAS APPROVED A PLAN FOR EXPEDIENT REPAIR OR REPLACEMENT OF THE FAILED UNIT AND (2) WITHIN 20 DAYS A REPORT HAS BEEN FILED WITH THE COMMISSION OUTLINING THE FINDINGS, CONCLUSIONS AND RECOMMENDATIONS OF THE SAFETY COMMITTEE.

5.3.6 IF EITHER THE HEAD-END OR MAIN PLANT EXHAUST VENTILATION SYSTEM FAILS TO SUCH AN EXTENT THAT NORMAL FLOW OF FILTERED AIR IS NOT AVAILABLE IN THE PARTICULAR SYSTEM, ALL PLANT OPERATIONS EXCEPT VENTILATION SYSTEM OPERATION SHALL BE SHUT DOWN IN AN ORDERLY MANNER.

5.3.7 THE INSTRUMENTATION MONITORING THE DIFFERENTIAL PRESSURE ACROSS THE OPERATING FINAL ABSOLUTE FILTER IN THE HEAD-END VENTILATION SYSTEM SHALL BE OPERABLE DURING OPERATION OF THE FILTER. IF THIS INSTRUMENTATION SHOULD BECOME INOPERABLE, OPERATION OF THE FILTER CAN CONTINUE FOR 10 DAYS, PROVIDED THAT THE INSTRUMENTATION MONITORING THE DIFFERENTIAL PRESSURE ACROSS THE FILTER BANK IS OPERABLE.

FILTER	PRIMARY INSTRUMENTS	ACCEPTABLE ALTERNATES
15T-73	15PDR-45 15PDAH-34	15PDR-44 15PDAH-32
15T-73A	15PDR-45A 15PDAH-34A	15PDR-44A 15PDAH-32A

THE INSTRUMENTATION MONITORING THE DIFFERENTIAL PRESSURE ACROSS THE OPERATING ABSOLUTE FILTERS IN THE MAIN VENTILATION SYSTEM SHALL BE OPERABLE DURING OPERATION OF THE FILTER. IF THIS INSTRUMENTATION SHOULD BECOME INOPERABLE, OPERATION OF THE FILTER MAY CONTINUE FOR 10 DAYS, PROVIDED THAT THE INSTRUMENTATION MONITORING THE DIFFERENTIAL PRESSURE ACROSS THE ABSOLUTE FILTER PLUS ROUGHING FILTER IS OPERABLE.

FILTER	PRIMARY INSTRUMENTS	ALTERNATES
15T-49	15PDR-11 15PDAH-11	15PDR-10 15PDAH- 5
15T-49A	15PDR- 6 15PDAH- 6	15PDR- 5 15PDAH- 4

5.3.8 WASTE TANK OFF-GAS SYSTEM—IF A FILTER, BLOWER, OR A WASTE TANK OFF-GAS CONDENSER FAILS, THE WASTE TANK VENT SYSTEM SHALL BE SWITCHED TO THE APPROPRIATE OPERABLE STANDBY EQUIPMENT AND IMMEDIATE REPAIR SHALL BE INSTITUTED TO EXPEDITE THE RETURN OF THE FAILED EQUIPMENT TO SERVICE.

5.3.9 DOG-VOG BACKUP FILTERS—A DOG-VOG BACKUP FILTER SHALL BE OPERABLE AT ALL TIMES.

Basis

The primary control of airborne radioactivity in the plant is provided by four ventilation systems. This specification has been designed to assure the availability of sufficient ventilation capacity to provide

the proper pressure differentials between operating equipment and occupied areas so that plant personnel are protected from airborne activity, and to maintain in operation essential treatment systems for the removal of radioactivity from the ventilation streams prior to their release to the atmosphere.

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, reheater, a silver nitrate reactor, a cooler and a filter prior to blower discharge into the stack. The chemical scrubber removes about 90 percent of the iodine 131 and, if necessary, the chemical scrubber in combination with a silver nitrate reactor removes about 99.5 percent of the iodine 131. If the minimum operable equipment requirements are not met, the evolution of radioactivity is reduced by discontinuing the shearing operation and by cooling the dissolver, thereby slowing the reaction. This will reduce the fraction of the annual release limit (Technical Specification 4.1.1) expended during equipment inoperability.

The VOG system provides equipment for the treatment of the gaseous effluents from radioactive-containing process vessels throughout the plant. These gases are condensed, scrubbed, heated and filtered prior to blower discharge to the stack. If both VOG filter-blower units fail, operations may continue for a 24-hour period because confinement is afforded by the DOG-VOG backup filter and the Main Ventilation System, which exhausts the spaces where the vessels on the VOG system are located.

An absolute filter and a parallel alternate are installed in the common discharge line from the DOG and VOG exhausters to the stack. These filters provide in-depth protection against releases from a failed filter in either the DOG-VOG systems. Technical Specification 5.3.9 requires that one of the backup filters must be operable. Hence, the spare must be maintained in readiness.

The exhaust system for the plant includes two major system. Each system must be fully operable before operations are begun at the start of a campaign in the areas served by each system. Thus the HEV system must be fully operable before fuel is moved into the PMC at the start

of a new campaign and the Main Ventilation System must be fully operable before commencement of solvent extraction at the start of a campaign.

Partial failure of either system will not result in undue risk provided that prompt repair of defective units is effected. Plant operations may continue without a spare in the areas served by each system for a 10-day period. If repairs cannot be made within 10 days, and if the failed equipment is a major component, operation may continue while repairs or replacement is made under the provisions of 5.3.5.

A total failure of either system will require the orderly shutdown of plant operations. Plant operations may be resumed upon the restoration to service of one full section of the failed system since the condition then is the same as would result from a partial failure during a campaign.

The Main Ventilation System controls the passage of air through the process areas beyond the Chemical Process Cell from clean areas to successively more contaminated areas and filters exhausted air prior to discharge up the stack. To provide for continuous operation a standby filter-exhauster unit is provided. If one of the filter banks becomes inoperable, plant operations may continue for a 10 day period without a spare while the filter bank is being returned to service. Ten days permit filter replacement without upsetting maintenance programs, yet entails minimal risk of being without spare. If both exhauster-filter systems are inoperative at the same time, an orderly shutdown must be initiated.

The Head End Ventilation System controls air flows for the mechanical processing areas, the Chemical Process Cell and the Manipulator Repair and Decontamination Building. Like the Main Ventilation System, the Head End Ventilation System includes independent filter systems and blowers as alternates. The HEV system has more flexibility since either filter bank can be operated with either the electric-driven or the propane-driven blower. The HEV system includes an even greater defense in depth since double absolute filtration is used. The operability requirements for HEV are similar to those established for the Main Ventilation System. Should the HEV system become totally inoperable, an orderly shutdown must be initiated.

Because of the potentially large inventories of radioactivity on the Main Ventilation filters and the Head End Ventilation filter, redundant instrumentation is provided. In addition to the stack monitor, performance of the operating filter is monitored by dual differential pressure recorders and high differential pressure alarms. The primary set of instruments monitors the absolute filters which are the final barrier to radioactivity release and the alternate set monitors the entire filter stage.

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

The consequences 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 Analyses. In showing that the consequences of events discussed in paragraphs 7.30 through 7.36 are acceptable within the guidelines of 10 CFR Part 100, reliance was placed on the absolute filters being effective.

6.4 EMERGENCY UTILITY EQUIPMENT

Applicability

This specification applies to the surveillance requirements for the standby utility equipment.

Objective

To assure that standby equipment necessary for continued facility operations is maintained in an operative 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 MONTHS TO DETERMINE THE AUTOMATIC START CAPABILITY AND THE 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. THE NEXT CAMPAIGN SHALL NOT START UNTIL THE EQUIPMENT IS RETURNED TO SERVICE.

6.4.2 THE EQUIPMENT LISTED BELOW SHALL BE OPERATED AT LEAST ONCE EVERY THREE MONTHS TO DETERMINE PERFORMANCE FOR AT LEAST ONE HOUR UNDER AN OPERATING LOAD. ANY DEFICIENCIES WILL BE CORRECTED AS SPECIFIED IN 6.4.1.

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

6.4.3 THE QUANTITY OF DIESEL FUEL IN THE OIL STORAGE TANK (31D-2) SHALL, DURING NORMAL OPERATIONS BE MAINTAINED AT 8,000 GALLONS OR MORE OR DIESEL FUEL. IF USED DURING ABNORMAL CONDITIONS, OPERATIONS MAY CONTINUE AS LONG AS A VOLUME OF 4,000 GALLONS OR GREATER IS MAINTAINED. A VOLUME LESS THAN THIS LATTER VALUE WILL REQUIRE THE COMMENCEMENT OF PLANT SHUTDOWN WHICH SHALL CONTINUE UNLESS THE PRIMARY FUEL (NATURAL GAS) SERVICE IS RESTORED OR THE FUEL OIL SUPPLY IS REPLENISHED.

6.4.4 THE AUXILIARY HEV BLOWER (15F-21) AND MAIN PLANT SPARE EXHAUST BLOWER (15K-10A) SHALL BE OPERATED AT LEAST ONCE EVERY THREE MONTHS TO DETERMINE THE AUTOMATIC START CAPABILITY AND THE PERFORMANCE FOR AT LEAST ONE HOUR UNDER AN OPERATING LOAD.

6.4.5 THE QUANTITY OF PROPANE FUEL IN THE PROPANE STORAGE TANK (15D-3) SHALL, DURING NORMAL OPERATIONS, BE MAINTAINED AT 600 GALLONS OR MORE. IF USED DURING ABNORMAL CONDITIONS, OPERATIONS MAY CONTINUE AS LONG AS A VOLUME OF 300 GALLONS OR GREATER IS MAINTAINED. A VOLUME LESS THAN 300 GALLONS WILL REQUIRE COMMENCEMENT OF SHUTDOWN OF OPERATIONS WHICH SHALL CONTINUE UNLESS THE PROPANE SUPPLY IS REPLENISHED.

Basis

Standby equipment has been provided to enhance the reliability of utilities supplied to safety equipment and devices and for building ventilation. In order to insure satisfactory performance, the standby equipment is 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 at least 10 hours. A propane supply sufficient for 72 hours operation of the HEV auxiliary blower has been provided. Unless weather conditions are very severe, diesel fuel and/or propane can be replenished within 10 hours thereby permitting continued operations. The consequences of failing to meet the requirements of this specification is to reduce the level of confidence in the operability of the standby services.

6.5 AIR FILTRATION EQUIPMENT

Applicability

This specification applies to 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 ROUTINE SERVICE, WITH PARTICULATES OF AN APPROPRIATE SIZE TO ESTABLISH THAT THE INSTALLED FILTERS PROVIDE A COLLECTION EFFICIENCY OF AT LEAST 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
MAIN BUILDING	15T-49, 15T-49A
DOG-VOG BACKUP	6T-3, 6T-3A
HEAD END	15T-72, 15&-72A, 15T-73, 15T-73A

6.5.2 THE DIFFERENTIAL PRESSURE ACROSS THE FINAL OPERATING ABSOLUTE FILTERS (OR ABSOLUTE PLUS ROUGHING WHERE THE FILTERS ARE WITHIN THE SAME FRAME) SHALL BE RECORDED BY INSTRUMENTATION.

6.5.3 ALARMS SHALL BE PROVIDED TO ANNUNCIATE HIGH DIFFERENTIAL PRESSURE ON THE FINAL OPERATING ABSOLUTE FILTERS OF THE MAIN BUILDING, HEAD END WASTE TANK FARM AND DOG-VOG BACKUP SYSTEMS. THESE ALARMS SHALL BE SET TO ANNUNCIATE AT LESS THAN 75% OF THE GREATEST DIFFERENTIAL PRESSURE AT WHICH THAT FILTER HAS BEEN TESTED.

6.5.4 AN ALARM SHALL BE PROVIDED TO ANNUNCIATE LOW DIFFERENTIAL PRESSURE ON THE FINAL OPERATING ABSOLUTE FILTER IN THE MAIN BUILDING

VENTILATION SYSTEM AND IN THE HEAD END VENTILATION SYSTEM. AFTER THE OPERATING DIFFERENTIAL PRESSURE EXCEEDS 5 INCHES W.G., THIS ALARM SHALL BE SET WITHIN 3 INCHES W.G. OF THE OPERATING DIFFERENTIAL PRESSURE AND SHALL BE ADJUSTED WEEKLY.

Basis

The final filters in the ventilation and off-gas systems are the primary protection against the discharge of particulate radioactivity during normal operations and particularly under accident conditions. These "absolute" filters are certified by the manufacturer to capture at least 99.97% of all particles 0.3 microns in diameter or larger. The filters are tested after insatllation to detect bypass leakage as well as filter damage that may have occurred during storage and handling. Subsequent tests are not necessary as installed process instruments and the stack radioactivity monitors will indicate filter integrity and performance after the filter is placed on stream.

These tests are conducted by putting the filter-blower in service, introducing a known amount of dioctylphthalate (DOP) 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 Specification 6.5.1 is to risk the introduction of a faulty filter. In such an event, the stack monitor would quickly show up the existence of the fault and corrective action would be taken.

Specification 6.5.2 requires that the differential pressure (Δp) across the final filters in the off-gas and ventilation systems be recorded by instrumentation in the Plant Control Room or the Waste Tank Farm shelter. The differential pressure serves as an indicator of filter performance.

Alarms which annunciate high differential pressure are provided, per Specification 6.5.3, on the absolute filters which are the final containment of radioactivity. These alarms are set to annunciate at about 90% of the maximum operating differential pressure allowed by Specification 4.14. This should provide the operator with sufficient time to switch to the alternate filter; however, the differential pressure may exceed the maximum operating differential pressure limit as a transient condition during removal from service.

Low differential pressure alarms are provided for the Main Building Ventilation and Head End Ventilation filters. Initially the low Δp alarm is set at about 1 inch W.G., i.e. at about 50% of the Δp experienced with a clean filter. After the Δp across the operating filter reaches 5 inches W.G., the alarm is adjusted weekly to be within 3 inches W.G. of the operating differential pressure. The purpose of these alarms is to increase the potential for detecting a failed absolute filter.

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through an air-washer and heater which the new head end vent system does not have. The elimination of the water spray air-washer and air heater is not expected to present a problem in the new ventilation system since materials of construction were selected to be resistant to moisture and nitric acid vapors which may be generated in the chemical processing cell. The new system provides double absolute filtration versus the washer and single absolute filtration in the existing system. The elimination of the washer and use of a more efficient first stage absolute filter in this system should increase the filtration efficiency for small (< 1 micron) particles. The existing technical specification which limits the amount of activity which can be released as a particulate from the stack is not changed.

The technical specifications which cover the operation and surveillance requirements for filter systems have been modified to include the new head end ventilation system and are consistent with the current specifications as they pertain to DOP testing of absolute filters and instrument surveillance of differential pressure on absolute filters.

On September 4, 1968, NFS experienced the complete rupture of a final absolute filter in the main ventilation train. After reviewing this matter, NFS concluded that the failure was due to a minimal strength filter. The filter was operating at less than 80% of the design pressure. As a result of the filter failure, quantities of filter media were drawn through the blower and dispersed out the stack. Subsequent to the September 1968 failure and prior to resuming operations following this failure, NFS installed a screen upstream of the main ventilation blower to trap filter media and prevent dispersal in case of filter rupture.

The HEV system does not have a screen downstream of the final absolute filter to trap possible filter media prior to disperal out the stack. A screen is, however, installed between the two banks of absolute filters to trap filter media from possible failure of the first absolute filter. The first absolute filter will contain the bulk of the particulate activity. The final filters, which are not backed up by a screen, as expected have a low inventory of particulate activity. NFS will supply information on the particulate loading of the final filter based on 6 and 12-month operating experience to justify not using a screen after the final absolute filter.

2. <u>Personnel Exposure</u>

NFS, by Supplement 2, gives anticipated average exposure rates for performing filter changes in the head end ventilation system. They

state that for the yearly operation of the system the cumulative annual exposure will be 1.3 man-rems with this exposure spread over 10-20 individuals. The exposure estimates are based on four pre-filter changes per year, two roughing filter changes per year and one change per year of the first absolute filters. NFS predicts that the filters in final absolute filter stages will have a service life of several years. The 1.3 man-rems per year exposure according to NFS represents an anticipated reduction of about 250 man-rem exposure per year.

The new system will take some activity load off the main system and the incorporation of remote changing of roughing and prefilters in the HEV system will therefore reduce the total plant personnel exposure. Operation of the HEV system will not eliminate personnel exposure resulting from main ventilation filter changing.

The analysis which was submitted by NFS in support of the 1.3 man-rem per year states that two minutes will be required to unclamp 15 absolute filters located in a glove box. NFS's time and motion studies indicate that four pre-filters can be changed in 92 minutes, 8 roughing filters can be in 105 minutes, and 15 absolute filters can be changed in 147 minutes. The operations which are involved include glove port aisle inspection, change room preparation, removal, transportation, and replacement.

NFS will be required to submit information on operating experience at intervals of 6 and 12 months after this approval.

On a qualitative basis personnel exposure should be reduced from that experienced from past operation of the main ventilation system.

3. MSM Sump Tank

A 1500 gallon waste tank has been provided to accumulate drainage from the master slave manipulator repair and decontamination shop, the filter housing area and the crane room. The tank is buried in the silty-till adjacent to the MSM shop. The activity in the solutions transferred to the sump tank are not accurately known but are stated by NFS to be low. The sump tank is stainless steel, 1/4" plate, welded construction and all welds have been radiographed.

There is no positive means of checking the integrity of the tank for leakage. NFS has installed a surveillance well from ground surface to the concrete pad foundation of the tank. The area around the bottom of the tank is packed with gravel to assure that tank leakage would be found by sampling the well for activity. A water level will be

maintained in the well to permit sampling and to help verify that solutions cannot flow from the area around the tank. This method of tank surveillance will be used for six months, after which time the system will be reevaluated.

Burial of the sump tank in the silty-till, which is found in the NFS area, restricts the possible flow of solutions away from the tank and the ion-exchange properties of the silty-till would also limit activity movement for possible limited periods of uncontrolled and detected tank leakage.

4. NATURAL PHENOMENA

The head end ventilation system was constructed to meet Seismic Zone III standards of the Uniform Building Code. While this was acceptable at the time the Plant was constructed it does not conform with the requirements currently imposed for aseismic design. In this connection NFS has assumed that in the event of earthquake or tornado the facility would fail.

The HEV building is constructed with the filter areas below grade within a reinforced concrete structure. The above ground structure serves as an airlock and crane room for filter removal.

The existing main ventilation system provides emergency back-up ventilation for those areas in the existing facility that are served by the HEV system.

Assuming that the HEV structure will fail under severe earthquake, the radioactivity would be contained under the rubble and no significant release would result. Location of the contaminated filters below grade reduces the possibility of contamination being dispersed by local wind conditions.

In the event of a tornado it is assumed that the above grade structure will be carried away and the access plugs to the filters will vent with possible radioactive material release. NFS performed an analysis of the effects of a postulated tornado which released 50% of the radioactive material contained by the filters. We are unable to confirm the NFS evaluation for the release of activity to unrestricted areas.

Our independent evaluation shows that the NFS postulated release of activity by a tornado would result in a worst case, maximum point, ground level concentration equivalent to 35 days exposure at the concentration limit for plutonium in unrestricted areas. The worst case calculation is based on the release of 50 percent (1.3 gms) of the

plutonium contained in the filter system. The isotopic composition of the plutonium was assumed to be that of irradiated power reactor fuels.

Our independent evaluation of the NFS estimated release of 62 curies of Cs-137 gives a maximum point ground level concentration equivalent to a one-day exposure at the concentration limit for Cs-137 in unrestricted areas.

CONCLUSION

Based on our review of the proposed installation, we conclude that it does not present significant hazard considerations not described or implicit in NFS's "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the new head-end ventilation system.

Approval of the attached technical specifications incorporating the operation of new head-end ventilation system is recommended.

Signed:

C. W. Nilsen
Irradiated Fuels Branch

Signed:

W. A. Nixon
Irradiated Fuels Branch

Approved:

R. B. Chitwood, Chief
Irradiated Fuels Branch
Division of Materials Licensing

CHANGE NO. 13

111975 JRC BEK TRW RTS

NFS

NUCLEAR FUEL SERVICES, INC.

6000 EXECUTIVE BOULEVARD, SUITE 600
ROCKVILLE, MARYLAND 20852
AREA CODE 301 · TELEPHONE 424-1750

February 11, 1971

Mr. Lyall E. Johnson
Acting Director of Materials Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

RE: Change No. 13
Technical Specification

Dear Mr. Johnson:

This will acknowledge receipt of your letter of January 26, 1971 concerning Change No. 13 in Technical Specification for the West Valley Plant. The suggested change does create practical problems in connection with Revised Technical Specification 5.1.4, Table 5.1, Section B. XXIII and Section E. We believe that it should be possible to resolve these problems through further discussion, and therefore, lest these sections of the Technical Specifications be finalized without further discussion, we request a hearing pursuant to 10 CFR Part 2, Rules of Practice limited to these two sections of the proposed Rule. To repeat, we hope that the problems, which represent a minor portion of the total impact of the revised Technical Specification, can be resolved through informal discussion.

In addition, we call to your attention two provisions of the Revised Technical Specification, which do not create serious problems but may not be realistic.

Specification 5.1.4, Table 5.1, Section C. III Fish Samples

The requirement for analysis of nine fish twice a year seems excessive, since we believe that the analysis of one or two fish per year would achieve the same results. Certainly an analysis of six fish would be more than adequate. The method of determining "geometric deviation" should be clarified.

RECEIVED
FEB 16 1971
J. P. DUCKWORTH

Section C. IV Deer Samples

It is entirely possible that there might be no deer on site during the hunting season or that the hunting skills of NFS personnel might be insufficient to enable them to take a deer for sampling.

We would appreciate some clarification on certain items. These concern the method of determining krypton in gaseous effluents and the frequency of analysis of particulates to determine isotopic ratios.

A detailed statement of the two main problems encountered in connection with Section B. XXIII and Section E, where we have requested a hearing is set forth below.

Revised Technical Specification 5.1.4, Table 5.1, Section B. XXIII

We respectfully request that this provision be deleted primarily because it serves no useful purpose, and the data accumulated under B. XXII can be utilized to achieve a more meaningful estimate of exposure in the various sectors.

In order to make the determination required under this section, it would be necessary for NFS to determine the meteorological conditions during each dissolver run and then make a separate calculation for each run. This calculation is completely different from that required in the previous Section XXII and is extremely difficult to accomplish.

The result of the calculation in each instance would be a figure reflecting an instantaneous exposure which might hold for a few minutes or a few hours, but which would be of no value in calculating average exposures. The limits of allowable MPC set up by 10 CFR 20, Appendix B are based on average exposures over a period of a year as recommended by the Federal Radiation Council and as recognized in the Regulations. There are no methods of determining whether or not an instantaneous exposure at the expected low levels of radioactivity is hazardous.

A more meaningful estimate of exposure in the various sectors can be made from the data accumulated under the previous Section XXII.

Technical Specification 5.1.4, Table 5.1, Section E

The U. S. Public Health Service in their Report BRH/NERHL 70-1 concluded that the exposure to the public as defined in 10 CFR 20.106 and Appendix B, Table II is completely negligible. To the best of our knowledge, during the entire operations of NFS, there has never been any "likelihood of public intakes in excess of 3% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20". We believe that this provision would be applicable only in the event of an unusual incident which would have other effects requiring NFS to be in touch with the AEC.

February 11, 1971

At the 3% level of exposure, the FRC merely requires routine sampling to determine that the contamination level is indeed low. If the level rose from 3% to, say 4%, the action required of NFS should be quite different from a situation where the level rose from 3% to 33%. While the requirements of the revised Specification - "resultant exposure to individuals and to population groups, and assumptions upon which estimates are based," - are probably sufficiently vague to cover all contingencies, their ambiguity could be a source of unnecessary arguments.

We, therefore, suggest that the Section be deleted:

a. Because the situation visualized is unlikely to arise;

b. If the situation did arise, it would be a relatively minor result of an incident with other effects, which would require a course of action approved by the Commission.

NFS is not especially well qualified in the area of public exposures, and any judgements as to public exposure that it might make could be clouded by its position as a vested interest. It would seem desirable to have such judgements come from an independent source such as the Public Health Service. Their work in this area is above reproach, and their surveillance of the West Valley Plant effluents is continuing.

If the Section is not deleted, it might be reworded as follows:

E. "If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 33% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, the Commission should be notified immediately in order that it may determine what further action is required.

Respectfully submitted,

E. D. North

E. D. North, Director
Technical Administration

EDN:kac

cc: B. G. Bechhoefer

COPY

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

January 26, 1971

Docket No. 50-201

Nuclear Fuel Services, Inc.
ATTN: Mr. Robert N. Miller, President
6000 Executive Boulevard
Suite 600
Rockville, Maryland 20852

and

New York Atomic and Space Development
Authority
ATTN: Mr. James Cline, General Manager
230 Park Avenue
New York, New York 10017

Change No. 13
License No. CSF-1

Gentlemen:

This is in response to a letter from Nuclear Fuel Services, Inc. (NFS) dated June 29, 1970, in which was submitted a proposal for an expanded environmental program. We subsequently discussed this proposal with Dr. E. D. North.

As a consequence of our review of the information contained in the NFS proposal and the discussions with Dr. North, we have determined to revise the Technical Specifications of License No. CSF-1 to incorporate an expanded environmental program. This revision to Section 5.1 of the Technical Specifications is contained in the attached Change NO. 13.

On the basis of our review, we have concluded that the change does not present significant safety hazards considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, pursuant to Sections 50.36 and 50.59 of 10 CFR Part 50, Change No. 13 in the Technical Specifications of License No. CSF-1, as set forth on the enclosed pages 44 through 46f of revised Technical Specification 5.1, is authorized.

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It should be noted that Change No. 13 does not reflect chances which may be necessary in connection with the operation of the low-level liquid waste treatment plant now being constructed.

Pursuant to 10 CFR Part 2, Rules of Practice, you may request a hearing with respect to all or any part of this change in the Technical Specifications within twenty (20) days from the date of this letter.

FOR THE ATOMIC ENERGY COMMISSION

(Original signed by
Lyall Johnson)

Lyall Johnson
Acting Director

Enclosure:
Change No. 13,
Rev. pp. 44 - 46f

cc: Mr. B. G. Bechhoefer w/encl
Mr. O. M. Ruebhausen w/encl
Mr. J. P. Duckworth w/encl

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5.1 EFFLUENT AND ENVIRONMENTAL MONITORING

Applicability

This specification applies to the sampling and analysis of gaseous and liquid plant effluents and to environmental monitoring.

Objective

To establish sampling points, sampling frequency and sample analytical requirements for gaseous and liquid plant effluents and to establish an environmental monitoring program and reporting requirement.

Specification

5.1.1 WHENEVER LIQUID IS DISCHARGED FROM THE STORAGE LAGOONS, A REPRESENTATIVE SAMPLE SHALL BE WITHDRAWN BY THE CATTARAUGUS CREEK CONTINUOUS SAMPLER. AT LEAST EVERY SEVEN DAYS, A COMPOSITE OF THIS SAMPLE SHALL BE ANALYZED FOR TRITIUM, GROSS ALPHA AND GROSS BETA RADIOACTIVITY.

IF THE CONTINUOUS SAMPLER BECOMES INOPERATIVE, UP TO 2,000,000 GALLONS MAY BE DISCHARGED FROM THE LAGOONS IN ANY ONE-YEAR PERIOD PROVIDED CATTARAUGUS CREEK SAMPLES ARE COLLECTED NEAR THE CONTINUOUS SAMPLER LOCATION FOR EACH LAGOON DISCHARGE OR FOR EACH 100,000 GALLONS OF DISCHARGE, WHICHEVER RESULTS IN THE GREATER SAMPLING FREQUENCY. SAMPLES SO COLLECTED SHALL BE INDIVIDUALLY ANALYZED FOR TRITIUM, GROSS ALPHA AND GROSS BETA RADIOACTIVITY.

5.1.2 A CONTINUOUS STACK SAMPLER SHALL BE USED TO COLLECT SAMPLES OF STACK GAS. THE SAMPLES SHALL BE ANALYZED AT LEAST EVERY SEVEN DAYS FOR IODINE 131 AND PARTICULATE RADIOACTIVITY. IF THE STACK SAMPLER BECOMES INOPERATIVE, IMMEDIATE REPAIR SHALL BE INSTITUTED TO RETURN THE SAMPLER TO SERVICE.

IN ADDITION TO THE WEEKLY SAMPLES OBTAINED FROM THE STACK SAMPLE, THE I-131, Kr-85 AND PARTICULATE RADIOACTIVITY IN THE STACK GAS SHALL BE CONTINUOUSLY MEASURED BY A STACK MONITOR.

(Change No. 13)

IF THE STACK SAMPLER BECOMES INOPERATIVE, IMMEDIATE REPAIR SHALL BE INSTITUTED TO RETURN THE UNIT TO SERVICE, AND THE STACK MONITOR SHALL BE USED IN THE INTERIM TO DETERMINE I-131 AND PARTICULATE RELEASES.

IF THE STACK MONITOR FAILS, A REPRESENTATIVE SAMPLE OF STACK GAS SHALL BE COLLECTED EACH SHIFT AND USED TO DETERMINE I-131 AND PARTICULATE RADIOACTIVITY. THE Kr-85 CONTENT OF STACK GASES SHALL BE CALCULATED FROM FUEL BURNUP DATA DURING PERIODS WHEN THE STACK MONITOR IS INOPERATIVE.

5.1.3 THE FOLLOWING INFORMATION, DETERMINED FROM THE SAMPLES TAKEN AS REQUIRED BY 5.1.1 AND 5.1.2, SHALL BE INCLUDED IN THE QUARTERLY OPERATING REPORT:

A) GROSS CURIES OF ALPHA ACTIVITY DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

B) GROSS CURIES OF BETA ACTIVITY DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

C) CURIES OF TRITIUM DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

D) CURIES OF SR-90 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

E) CURIES OF I-129 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

F) CURIES OF PARTICULATES DISCHARGED VIA THE STACK FOR EACH MONTH.

G) THE PERCENT OF THE MONTHLY TECHNICAL SPECIFICATION LIMIT FOR DISCHARGE OF PARTICULATES VIA THE STACK.

H) THE MAXIMUM PERCENT OF THE KR-85 DAILY TECHNICAL SPECIFICATION LIMIT IN THE STACK EFFLUENT FOR EACH MONTH.

I) THE CUMULATIVE PERCENT OF THE I-131 YEARLY TECHNICAL SPECIFICATION LIMIT IN THE STACK EFFLUENT REPORTED ON A MONTHLY BASIS.

Basis

Effluent Monitoring

(Changes No. 13)

Samples of water from Cattaraugus Creek are taken with a continuous sampler located about one-half mile downstream from the confluence of Cattaraugus and Buttermilk Creeks. Portions of each weekly sample are composited to produce a sample representative of one month's discharge. The weekly samples are analyzed for gross alpha, gross beta and tritium. The composite is analyzed for gross alpha, gross beta and strontium 90.

A stream gage located near the sampling station is used to determine total flow in Cattaraugus Creek. Flow from the lagoon is determined by a calibrated weir located on the discharge line. The flow ratio is used in conjunction with the iodine 129 analysis of lagoon water to calculate the iodine 129 content in Cattaraugus Creek.

Gaseous plant effluents are sampled in the plant stack. The stack sampler contains a filter to collect particulates and an activated charcoal filter to collect iodine 131. Samples are removed from the stack sampler at least each week and analyzed.

The stack monitor is used to continuously determine the particulate, iodine 131 and krypton 85 radioactivity in the stack air and to alert operators if pre-set limits are exceeded. The particulate sampler is changed every 8 hours and will alarm if the accumulation of particulate radioactivity over an 8-hour period exceeds that which would occur if particulates were being discharged over the same time limit at the limit set by Technical Specification 4.1. The iodine 131 monitor, a carbon filler paper absorber, is changed on a weekly cycle. The instrument is set to alarm if, over a week's period, the counting rate increases 10,000 counts/minute. This is far below the increase in count rate (about 400,000 counts/minute) which would occur if iodine were released at the technical specification limit for one week. The krypton 85 monitor is set to alarm if the counting rate exceeds five times the background rate. The alarm point is not, in this case, directly related to allowable release rates. The instrument will alarm only during periods of high krypton 85 discharge, for example during dissolution of high burn-up fuels.

5.1.4 THE LICENSEE SHALL CONDUCT THE ENVIRONMENTAL MONITORING PROGRAM GIVEN IN TABLE 5.1 AND SHALL REPORT THE RESULTS OBTAINED AT THE END OF EACH SIX-MONTH PERIOD BEGINNING FROM JANUARY 1, 1971. SUCH REPORTS WILL BE DUE WITHIN 60 DAYS AFTER THE END OF EACH REPORTING PERIOD. IF QUANTITIES OF RADIOACTIVE MATERIAL RELEASED DURING THE REPORTING PERIOD ARE UNUSUAL FOR NORMAL PLANT OPERATIONS, INCLUDING EXPECTED OPERATIONAL OCCURRENCES, THE REPORT SHALL COVER THIS SPECIFICALLY.

(Changes No. 13)

TABLE 5.1

EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

A. The licensee shall determine for each month, based on samples taken at the indicated location and other information collected during the month, the following:

i Gross curies of beta radioactivity other than tritium released at effluent weir.

ii Gross curies of alpha radioactivity released at the effluent weir.

iii Curies of tritium released at effluent weir.

iv Curies of strontium 90 released at effluent weir.

v Curies of cesium 134 released at effluent weir.

vi Curies of cesium 137 released at effluent weir.

vii Curies of ruthenium 106 and rhodium 106 released at effluent weir.

viii Volume of water released at effluent weir.

ix Volume of water flow through site in Cattaraugus Creek.

x Curies of beta emitting particulates released via the stack.

xi Curies of alpha emitting particulates released via the stack.

xii Curies of iodine 131 released via the stack.

xiii Curies per cubic meter gross beta radioactivity of particulates (average and maximum) collected by each of the 3 site perimeter samplers on filter paper.

xiv Curies per cubic meter gross alpha radioactivity of particulates (average and maximum) collected by each of the 3 site perimeter samplers on filter paper.

(Change No. 13)

B. The licensee shall determine each quarter, based on samples and other information collected during the quarter, the following:

i Identity of principal radionuclides whose presence can be determined by gamma spectroscopy of a sample (one per calendar quarter) of Buttermilk Creek bottom silt collected at the Thomas Corner Road bridge.

ii Microcuries per gram gross beta radioactivity of a sample (one per calendar quarter) of Buttermilk Creek bottom silt collected at the Thomas Corner Road bridge.

iii Microcuries per gram gross alpha radioactivity of a sample (one per calendar quarter) of Buttermilk Creek bottom silt collected at the Thomas Corner Road bridge.

iv Microcuries tritium per milliliter of water collected once per quarter at the Buttermilk Creek silt sampling location during a normal liquid effluent release from the plant.

v Microcuries gross beta radioactivity per milliliter of water collected once per quarter at the Buttermilk Creek silt sampling location during a normal liquid effluent release from the plant.

vi Microcuries gross alpha radioactivity per milliliter of water collected once per quarter at the Buttermilk Creek silt sampling location during a normal effluent release from the plant.

vii Curies of strontium 89 released at the effluent weir based on an analysis of a composite of the liquid released during the last month of the quarter.

viii Curies of zirconium 95 and niobium 95 released at the effluent weir based on an analysis of a composite of the liquid released during the last month of the quarter.

ix Curies of iodine 129 released at effluent weir.

x A quantitative analysis of alpha emitting components in a quarterly composite collected at the effluent weir.

(Change No. 13)

xi The exposure, in megawatts days, of fuel dissolved during the quarter.

xii The curies of Krypton 85 released via the stack as determined using the stack monitor.

xiii Curies of tritium released via the stack.

xiv Curies of iodine 129 released via the stack.

xv Curies of strontium 89 released via the stack.

xvi Curies of strontium 90 released via the stack.

xvii Curies off ruthenium 106 released via the stack.

xviii Curies of cesium 134 released via the stack.

xix Curies of cesium 137 released via the stack.

xx Microcuries per milliliter tritium (average) in precipitation collected at each of the 3 site perimeter sampling stations.

xxi Milliroentgen of radiation (including background) accrued each quarter at each of the 3 site perimeter sampling stations.

xxii The maximum value at ground level offsite in each $\pi/8$ sector about the stack of the X/Q integrated for each seasonal quarter from hourly averages of continuously recorded meteorological data. The X/Q shall be determined using equation 3.144 given on page 113 of Meteorology and Atomic Energy (1968). Seasonal quarters are defined as:

Spring	March through May
Summer	June through August
Fall	September through November
Winter	December through February

xxiii The curie seconds per cubic meter maximum exposure of Kr-85 integrated for the reported quarter at ground level offsite in each $\pi/8$ sector about the stack, computed from meteorological data recorded during emission of the respective increments of Kr-85 that register above background on the Kr-85 stack gas monitor.

(Change No. 13)

C. The licensee shall determine the following information according to the schedule given:

i If, during any week from April to October, inclusive, the release of iodine 131 from the stack exceeds 30 millicuries, the licensee shall determine the microcuries per milliliter of iodine 131 in samples of milk from farms within 2-1/2 miles of the plant in each direction expected to receive the release. One sample shall be collected each week for the four weeks following the release. Each sample shall be a composite of one day's production from cows which are on pasture.

ii During August of each year the licensee shall collect a milk sample from a farm within 2-1/2 miles of the plant in the north-west sector and a milk sample from a farm within 2-1/2 miles of the plant in the north-east sector. The samples shall be composites of one day's production from cows which are on pasture. Each sample shall be analyzed separately to determine:

a) Microcuries iodine 129 per milliliter

b) Microcuries strontium 90 per milliliter

c) Microcuries cesium 134 per milliliter

d) Microcuries cesium 137 per milliliter

iii During the second and third quarters of each year, licensee shall take fish samples from Cattaraugus Creek between the Springville hydroelectric dam and the Cattaraugus Creek-Buttermilk Creek confluence. One sample shall be taken during each of the two quarters. A sample shall consist of at least 9 fish, each at least six inches long. Each fish in each sample shall be analyzed to determine:

a) Median and geometric deviation of cesium 134 microcuries per kilogram of flesh.

b) Median and geometric deviation of cesium 137 microcuries per kilogram of flesh.

c) Median and geometric deviation of strontium 90 microcuries per kilogram of flesh.

(Change No. 13)

d) Median and geometric deviation of strontium 90 microcuries per kilogram of skeleton.

iv Once per year the licensee shall determine the following from a deer taken onsite during the hunting season and taken, preferably, north of the plant:

a) Microcuries cesium 134 per kilogram of flesh.

b) Microcuries cesium 137 per kilogram of flesh.

c) Microcuries strontium 90 per kilogram of flesh.

d) Microcuries strontium 89 per kilogram of skeleton.

e) Microcuries strontium 90 per kilogram of skeleton.

D. The environmental reports to be prepared by the licensee shall contain descriptive material covering the environmental surveys performed during the reporting period including information on:

i The number and types of samples taken; e.g., air, surface water, soil, fish.

ii The number and types of measurements made.

iii Locations of the sample points and monitoring stations.

iv The frequency of the surveys.

v A summary of survey results.

E. If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 3% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based should be provided.

F. If statistically significant variations in offsite environmental concentrations with time are observed, correlation of these results with effluent releases should be provided.

Basis

Environmental Monitoring

The environmental survey incorporates measurements to provide background data and measure possible plant effects. Samples collected at points where concentrations of effluents in the environment are expected to be greatest will be compared with samples collected concurrently at points expected to be essentially unaffected by station effluents. The latter samples will provide background measurements as a basis for distinguishing significant radioactivity introduced into the environment by the operation of the plant from that due to other sources such as nuclear detonations.

This schedule will assure that significant changes in the environmental radioactivity are sampled most frequently. Those which are less affected by transient changes but show long-term accumulations are sampled less frequently.

The monitoring program will be reviewed and re-evaluated periodically after data such as composition, quantity and frequency of releases, dilution factors and measured concentrations in food and other organisms (if any are observed) are obtained.

CHANGE NO. 14

NFS

NUCLEAR FUEL SERVICES, INC.

BOX 124 ··· WEST VALLEY, N.Y. 14171
AREA CODE 716 TELEPHONE 942-3235

July 12, 1971

Mr. Richard E. Cunningham, Acting Director
Division of Materials Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Cunningham:

Pursuant to Paragraph 50.59 of Title 10 Code of Federal Regulations, Nuclear Fuel Services, Inc. hereby requests a change in the Technical Specifications for Operating License CSF-1. The proposed changes will permit the NFS Processing Plant at West Valley, New York, to receive and store plutonium nitrate-uranyl nitrate solutions now being prepared at the NFS facility at Erwin, Tennessee. The safety evaluation of the proposed processing plan and the necessary amendments to the Technical Specifications will be the subject of an up-coming submission.

NFS believes that these changes, which are necessary for receipt and storage and which were discussed with personnel of the Irradiated Fuels Branch, do not contain any significant safety consideration not described or implicit in the Final Safety Analysis Report submitted under Docket 50-201; therefore, authorization of the changes is requested.

Very truly yours,

J. P. Duckworth

J. P. Duckworth
Plant Manager

JPD:cp
Enc.
cc: E. D. North
D. H. Shafer
B. G. Bechhoefer

PROPOSED TECHNICAL SPECIFICATION CHANGES FOR THE RECEIPT AND STORAGE OF $PU(NO_3)_4$-$UO_2(NO_3)_2$ AT THE WEST VALLEY PROCESSING PLANT

The solutions of plutonium nitrate and depleted uranyl nitrate will be packaged and transported to the NFS Fuel Processing Plant in containers approved by the USAEC Division of Material Licensing and Department of Transportation. It is anticipated that these containers will be either NFS 10 LI or ASDA 10 LI packages.

The packages will be stored in Product Packaging & Storage (PPS) Area of the Processing Plant which has previously been reviewed by the USAEC and approved for such storage by Technical Specification 4.10 of License CSF-1. The total number of containers in the PPS will not exceed 125; therefore, the possession limit for the facility as listed in Specification 3.1.2 need not be amended. The amendments that NFS believes necessary are shown below.

3.1.1 (new paragraph)

Source and/or special nuclear materials may also be received solutions in the following category:

Category	Fuel Compound	Fissile Concentration g/l
10	Plutonium Nitrate-Depleted Uranyl Nitrate	Not exceeding 250 grams fissile plutonium per liter

3.1.3 (new third paragraph)

Material Category 10 authorized in Specification 3.1.1 may be received in packages authorized by the USAEC and the Department of Transportation. Storage shall be in accordance with Technical Specification 4.10.1.2 until processing.

Bases (change)

Delete irradiated from second line.

NFS

NUCLEAR FUEL SERVICES, INC.

BOX 124 — WEST VALLEY, N.Y. 14171
AREA CODE 716 TELEPHONE 942-3235

July 27, 1971

Mr. R. E. Cunningham, Acting Director
Division of Materials Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Cunningham:

The following information is presented to supplement our request of July 12, 1971, for authorization to receive and store plutonium nitrate solutions at the West Valley Processing Plant. The NFS Plant Safety Committee has reviewed the proposed receipt and storage and believes that these actions can be accomplished in a safe manner using the plant's standard operating procedures that have been developed during the storage and shipment of more than 1700 kilograms of plutonium as nitrate solutions. The receipt and storage will be under the supervision of personnel possessing senior licenses; however, since these operations will be in geometrically safe containers, NFS does not believe the operators themselves should need specific AEC licenses. The shipping containers will be vented through filters during storage. Subsequent transfer to process will be accomplished by operators licensed by the USAEC for plutonium loadout and will be under standard operating procedures approved by the Plant Safety Committee.

Very truly yours,

J. P. Duckworth
Plant Manager

JPD:cp

cc: E. D. North
D. H. Shafer
B. G. Bechhoefer

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U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

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

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August 30, 1971

Docket No. 50-201

Change No. 14
License No. CSF-1

Nuclear Fuel Services, Inc.
ATTN: Mr. J. P. Duckworth, Plant Manager
Box 124
West Valley, New York 1417

And

New York Atomic and Space Development
Authority
ATTN: Mr. James Cline, General Manager
230 Park Avenue
New York, New York 10017

Gentlemen:

This refers to Nuclear Fuel Services' letters dated July 12, 1971 and July 27, 1971, requesting a change in the Technical Specifications of License No. CSF-1 to authorize receipt and storage of plutonium nitrate-uranyl nitrate solutions at the West Valley, New York, facility.

We reviewed the existing Technical Specifications and determined that changes in specification 3.1 and 4.10 would be necessary to authorize receipt and storage of plutonium nitrate-uranyl nitrate solution. In addition, we decided to take this opportunity to modify other portions of specifications 3.1, 4.10 and 6.1 in accordance with earlier NFS requests or as discussed with NFS personnel.

We have determined that the changes in the Technical Specifications, as set forth in Change No. 14, do not present significant hazard considerations not described or implicit in the NFS Final Safety Analysis Report and that there is reasonable assurance that the health and safety of the public will not be endangered.

A copy of the Safety Evaluation by the Division of Materials Licensing relating to Change No. 15 is enclosed.

Pursuant to Section 50.59 of 10 CFR Part 50, changes to Technical Specifications 3.1, 4.10 and 6.1 of Provisions Operating License CSF-1 (as set

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forth on enclosed pages 5, 6, 7, 8, 35, 36, 36a, 57 and 58) are authorized and all Technical Specifications so designated previously are rescinded.

FOR THE ATOMIC ENERGY COMMISSION

(Original signed by
Richard E. Cunningham)
Acting Director
Division of Materials Licensing

Enclosures:
1. Change no. 14, Rev. Pgs. 5, 6, 7, 8, 35, 36, 36a, 57 and 58
2. Safety Evaluation

cc: Mr. B. G. Bechhoefer, w/encl.
Mr. G. M. Ruebhausen, w/encl.
Dr. E. D. North, w/encl.
Mr. Robert N. Miller, w/encl.

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Material	Possession Limit	Form
Plutonium	62. grams	any
Plutonium	14. grams	sealed source
Plutonium-242	6. grams	any
Plutonium-238	1. gram	any
Neptunium-237	3.5×10^{-3} curie	any
Americium-241	1.1×10^{-9} curie	any
Americium-241	$6. \times 10^{-7}$ curie	sealed source
Thallium-204	$5. \times 10^{-6}$ curie	any
Cesium-137	$6. \times 10^{-5}$ curie	any
Cesium-137	33. curies	sealed source
Cesium-134	$6. \times 10^{-5}$ curie	any
Cerium-144	$3. \times 10^{-5}$ curie	any
Iodine-131	$3. \times 10^{-6}$ curie	any
Iodine-129	$6. \times 10^{-6}$ curie	any
Ruthenium-106	$5. \times 10^{-5}$ curie	any
Zirconium-95	$3. \times 10^{-5}$ curie	any
Strontium-90	$3. \times 10^{-3}$ curie	any
Strontium-85	$6. \times 10^{-3}$ curie	any
Krypton-85	$5. \times 10^{-3}$ curies	any
Zinc-65	$2. \times 10^{-5}$ curie	any
Cobalt-60	$5. \times 10^{-2}$ curie	any
Any byproduct material with atomic numbers from 3 to 85 inclusive	$3. \times 10^{-6}$ curie each	any

(Change No. 14)

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The consequence of using other than nuclearly safe systems for the storage of plutonium solutions is to provide a possibility for the occurrance of nuclear criticality in an unshielded area.

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SAFETY EVALUATION

by

THE DIVISION OF MATERIALS LICENSING

NUCLEAR FUEL SERVICES, INC.

DOCKET NO. 50-201

RECEIPT AND STORAGE OF PLUTONIUM SOLUTIONS

Authorization Requested

By letter dated July 12, 1971, as supplemented by a letter dated July 27, 1971, Nuclear Fuel Services, Inc. (NFS) requested a change in Technical Specification 3.1 of License No. CSF-1 to allow the receipt and storage of plutonium nitrate-uranyl nitrate solutions at the West Valley plant. After review of the existing Technical Specifications it was determined that changes should be made not only in Technical Specifications 3.1, Nuclear Fuel; but also in 4.10, Plutonium Solution storage and 6.1, Boron Raschig Rings. The modifications required in Technical Specifications 4.10 and 6.1 are similar to those requested by NFS by letter dated February 10, 1970 as part of an overall Technical Specification updating. Modifications to Technical Specification 6.1 were not necessary for receipt and storage of plutonium solutions at NFS, but were made because of the relationship between Specifications 4.10 and 6.1. Modifications to be made to Technical Specifications were discussed with NFS personnel, and their verbal agreement to the changes was obtained on July 26, 1971.

Authorization for receipt and storage of plutonium nitrate-uranyl nitrate solutions at the West Valley site has been granted by changes to Technical Specifications 3.1, Nuclear Fuel; and 4.10, Plutonium storage. Specification 6.1 has been modified to be consistent with Specification 4.10.

An additional change to Technical Specification 3.1, Nuclear Fuel, was made to make it clear that NFS may possess quantities of materials exempt under the provisions of 10 CFR Part 30.18 in addition to larger quantities of radioisotopes specifically listed in the Technical Specifications for use as standards, and for tests, measurements and calibrations.

Background

As a result of past fuel fabrication activities at NFS' Erwin, Tennessee, facility, plutonium-depleted uranium scrap which has accumulated is being processed. The processing steps at Erwin are limited to scrap dissolution in nitric acid, concentration, and packaging in AEC approved shipping containers. The plutonium nitrate-depleted uranyl nitrate solutions will be shipped to the NFS West Valley site and, at a later date, processed through the reprocessing plant for plutonium and uranium separation, purification and recovery. The present application for a change in Technical Specifications requests authorization only for receipt and storage of plutonium nitrate-uranyl nitrate solutions at West Valley. The safety evaluation for the proposed recovery operations at the West Valley plant and a request for changes in Technical Specifications to authorize the recovery operations will be the subject of a future submission by NFS.

Safety Considerations

1. General

Operations involved in the receipt and storage of plutonium nitrate-uranyl nitrate solutions at the West Valley plant have been reviewed and approved by the NFS plant safety committee. A standard operating procedure has been written and trained personnel will be used for the work. The safety considerations for the receipt and storage of the material are similar to those for the storage and shipment of plant produced plutonium nitrate solutions. Operations involved in receipt and storage of the solutions do not involve manipulation of controls as defined in 10 CFR Part 55 thus licensing of operators is not required.

2. Technical Specification 3.1, Nuclear Fuels

Category 10, plutonium nitrate-depleted uranyl nitrate, has been added to the list of materials which NFS may receive at the West Valley site. The concentration limit of 250 grams per liter of fissile plutonium corresponds to the limit placed on the shipping containers under the provisions of 10 CFR Part 71.

Section 3.1.3 of the specification has been modified to allow receipt and storage of category 10 material but to prevent transfer of the material from the shipping packages before AEC approval of NFS' processing plan for the material is granted.

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

September 24, 1971

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Docket No. 50-201

Nuclear Fuel Services, Inc.
ATTN: Mr. Robert N. Miller
President
6000 Executive Boulevard
Suite 600
Rockville, Maryland 20852

and

New York Atomic and Space
Development Authority
ATTN: Mr. James Cline
General Manager
230 Park Avenue
New York, New York 10017

Change No. 15
License No. CSF-1

Gentlemen:

This is in response to the January 14, 1971 letter from Nuclear Fuel Services, Inc. (NFS), requesting authorization to operate the new low level waste treatment plant. The new plant was described in a Final Safety Analysis Report submitted with the January 14 letter and in Supplement 1 submitted by letter dated April 2, 1971.

Based on our review of the information contained in the NFS submittals and discussions with the NFS staff and in accordance with the objective of reducing the quantity and concentrations of radioactivity in liquid effluent from the NFS facility, we have, in connection with the authorization sought, revised the Technical Specifications of License No. CSF-1 to incorporate Technical Specifications related to operation of the main plant ant to the low level liquid waste treatment plant and to discharge of effluents to the watershed.

In connection with the operation of the low level waste treatment plant the following interim reporting requirements shall apply?

1. A report every six months giving information on personnel radiation exposure associated with the operation of the new facility.

2. A report every six months giving information on the ruthenium decontamination factors achieved during operation of the new facility, and a description of studies or changes made by NFS to improve ruthenium removal or control capability.

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The time periods for the above listed reports shall start on the effective date of the revised Technical Specifications.

On the basis of our review, we have concluded that the changes in the waste treatment system will substantially reduce the quantity and concentrations of radioactivity in liquid effluent from the NFS facility. A copy of the Safety Evaluation by the Division of Materials Licensing of the new low level waste treatment plant is enclosed.

Accordingly, pursuant to 10 CFR Parts 2, 20 and 50, Change No. 15 in the Technical Specifications of License No. CSF-1, as set forth on the enclosed pages 15 through 16c, 42a, 44 through 46g, 63 and 72 revising previous Technical Specifications 4.2, 4.14, 5.1, 6.5 and 7.1 shall become effective twenty (20) days from the date of this letter.

Until Change No. 15 in the Technical Specifications of License No. CSF-1 becomes effective, test operation of the low level waste treatment plant may be conducted in accordance with the presently effective Technical Specifications.

Pursuant to the Commission's Rules of Practice in 10 CFR Part 2, you may request a hearing with respect to all or any part of this change in the Technical Specifications within twenty (20) days from the date of this letter.

FOR THE ATOMIC ENERGY COMMISSION

Richard E. Cunningham
Acting Director
Division of Materials Licensing

Enclosures:
1. Change No. 15 including
   Rev. pp. 15 through 16c, 42a,
2. Safety Evaluation

cc w/encl:
Mr. B. G. Bechhoefer
Mr. O. M. Ruebhausen
Mr. J. P. Duckworth

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4.2 RADIOACTIVITY CONTENT OF RELEASED LIQUID EFFLUENTS

APPLICABILITY

This is an interim specification which applies to radioactivity contained in liquid effluents from the site.

OBJECTIVE

To limit the concentrations of radioactivity in liquid effluents discharged from the site.

SPECIFICATION

4.2.1 ALL PROCESS STEAM CONDENSATE SHALL BE COLLECTED BATCHWISE AND SAMPLED FOR TOTAL ALPHA AND BETA ACTIVITY. THE CONDENSATE MAY BE DISCARDED WITHOUT PASSING THROUGH THE INTERCEPTOR, STORAGE LAGOON OR LOW LEVEL WASTE TREATMENT PLANT IF IT HAS BEEN DETERMINED THAT THE TOTAL ALPHA PLUS BETA ACTIVITY IS LESS THAN 3.0×10^{-7} MICROCURIES PER MILLILITER (EXCLUDING TRITIUM).

4.2.2 LAUNDRY WASTE SHALL BE COLLECTED BATCHWISE AND SAMPLED FOR TOTAL ALPHA AND BETA ACTIVITY. THIS WASTE MAY BE DISCHARGED TO THE LAGOONS DOWNSTREAM OF THE LOW LEVEL WASTE TREATMENT PLANT IF THE TOTAL ALPHA PLUS BETA ACTIVITY IS LESS THAN 1×10^{-4} MICROCURIES PER MILLILITER.

4.2.3 A. EXCEPT AS PROVIDED IN 4.2.3 B, ALL OTHER LIQUID WASTES CONTAINING RADIOACTIVITY OR WHICH HAVE BEEN IN CONTACT WITH EQUIPMENT CONTAINING RADIOACTIVITY SHALL BE COLLECTED BATCHWISE IN THE INTERCEPTORS, ANALYZED FOR TOTAL ALPHA PLUS BETA RADIOACTIVITY, AND TREATED IN THE LOW LEVEL WASTE TREATMENT PLANT.

B. IF THE TOTAL RADIOACTIVITY OF WASTE IN THE INTERCEPTOR IS GREATER THAN 5.0×10^{-3} MICROCURIES PER MILLILITER (EXCLUDING TRITIUM), THE WASTE SHALL BE TRANSFERRED BACK TO THE MAIN PLANT FOR FURTHER TREATMENT OR STORAGE.

4.2.4 IF DURING OPERATION OF THE WASTE TREATMENT PLANT, THE FLOCCULATION, CLARIFICATION, OR ION EXCHANGE SECTIONS OF

THE LOW LEVEL WASTE TREATMENT PLANT ARE BYPASSED, THE TREATED WATER SHALL BE RECYCLED TO THE FEED OF THE LOW LEVEL WASTE TREATMENT PLANT.

4.2.5 TREATED WASTE MAY BE DISCHARGED FROM THE LAGOON SYSTEM PROVIDED THAT THE RADIOACTIVE CESIUM AND STRONTIUM CONCENTRATIONS IN THE WASTE AT THE POINT OF DISCHARGE FROM THE LAGOON SYSTEM DO NOT EXCEED VALUES THAT SATISFY THE SUM OF THE FOLLOWING RATIOS:

$$\frac{C_{Cs-134}}{9 \times 10^{-6}} + \frac{C_{Cs-137}}{2 \times 10^{-5}} + \frac{C_{Sr-89}}{3 \times 10^{-6}} + \frac{C_{Sr-90}}{3 \times 10^{-7}} \leq 1$$

WHERE C = CONCENTRATION, IN MICROCURIES PER MILLILITER, OF THE INDICATED ISOTOPE IN THE TREATED WASTE.

IF THE CONCENTRATIONS OF CESIUM AND STRONTIUM RADIOISOTOPES PERMITTED BY THIS SPECIFICATION ARE EXCEEDED, THE LICENSEE SHALL:

(a) TAKE SUCH ACTION AS IS NECESSARY TO COME INTO PROMPT COMPLIANCE WITH THE TECHNICAL SPECIFICATIONS,

(b) MAKE AN INVESTIGATION TO IDENTIFY THE CAUSE OR CAUSES FOR SUCH LEVELS OF RADIOACTIVITY,

(c) DEFINE AND INITIATE A PROGRAM OF ACTION TO REDUCE SUCH LEVELS, AND

(d) REPORT THESE ACTIONS TO THE COMMISSION ON A TIMELY BASIS.

SUBJECT TO THE PROVISIONS OF 4.2.6, TRITIUM AND RUTHENIUM 106 MAY EXCEED, AT THE POINT OF DISCHARGE FROM THE LAGOON SYSTEM, THE RESPECTIVE CONCENTRATIONS LISTED FOR THESE NUCLIDES IN APPENDIX B, TABLE II, 10 CFR PART 20.

4.2.6 RADIOACTIVITY IN CATTARAUGUS CREEK RESULTING FROM THE DISCHARGE OF ALL LIQUIDS FROM THE SITE IS DETERMINED FROM WEEKLY COMPOSITE SAMPLES TAKEN IN ACCORDANCE WITH SPECIFICATION 5.1.1. IF THE RADIOACTIVITY IN CATTARAUGUS CREEK AS DETERMINED FROM SUCH SAMPLES(S) EXCEEDS:

(1) TWENTY PERCENT (20%) OF THE PRORATED CONCENTRATIONS LISTED IN APPENDIX B, TABLE II, 10 CFR PART 20 FOR ANY SAMPLE, OR

(2) TEN PERCENT (10%) OF THE PRORATED CONCENTRATIONS LISTED IN APPENDIX B, TABLE II, 10 CFR PART 20 AVERAGED OVER ANY QUARTERLY PERIOD,

THE LICENSEE SHALL:

(a) TAKE SUCH ACTION AS IS NECESSARY TO COME INTO PROMPT COMPLIANCE WITH THE TECHNICAL SPECIFICATIONS,

(b) MAKE AN INVESTIGATION TO IDENTIFY THE CAUSE OR CAUSES FOR SUCH LEVELS OF RADIOACTIVITY.

(c) DEFINE AND INITIATE A PROGRAM OF ACTION TO REDUCE SUCH LEVELS, AND

(d) REPORT THESE ACTIONS TO THE COMMISSION ON A TIMELY BASIS.

BASIS

These specifications limit:

the total radioactivity transferred to the storage lagoons from the main plant,

the cesium and strontium radioactivity in the waste discharged from the lagoon system after treatment in the low level liquid waste treatment plant and

the concentration of radioactivity in Cattaraugus Creek.

After treatment in the low level liquid waste treatment plant the dominant radioisotopes in the liquid effluent are expected to be ruthenium and tritium. The tritium is not affected by the treatment system and the ruthenium is not effectively removed by the present treatment system.

This is an interim specification in that the ultimate objective of liquid waste control is to reduce radioactivity discharges to as low a level as is practicable for all isotopes in liquids at the point of release from the lagoon systems. Additional treatment or other disposal methods may be needed in the future to meet the objective.

4.2.1 Process steam condensate is normally expected to be free of radioactivity. This effluent may be discharged to the environment if it has been determined that the total radioactivity content is less than the 10 CFR 20 limit for strontium-90,

one of the most restrictive isotopes which could be expected to be found in liquid wastes from reprocessing plants. The 10 CFR Part 20 limit for strontium-90 is ten to one hundred times smaller than the limit for other isotopes which would also be present in the condensate should a contaminating leak occur in a steam heating system. Thus use of the strontium-90 limit assures that the condensate released will always be below the 10 CFR 20 limit for any isotopes which might be present.

4.2.2 Laundry waste contains small quantities of radioactivity. It also contains phosphates and complexing agents which may interfere with the proper operation of the low level liquid waste treatment plant. This waste, therefore, is discharged downstream of the low level waste treatment plant.

The quantity of laundry waste is less than 10% of the total current low level liquid waste volume generated, and the activity level is about 5×10^{-5} microcuries per milliliter total beta. Both strontium and cesium are expected to be below their respective Part 20 MPC values in the undiluted waste. Experience indicates that the total beta activity carried by this stream is about 1 millicurie per day. This is predominantly ruthenium-106 with cesium-137 and strontium-90 amounting to about 18 and 4 microcuries per day respectively. Release of this waste without treatment will not substantially increase the total radioactivity release.

The laundry waste will be collected batchwise and sampled to assure that it does not exceed 1×10^{-4} microcuries per milliliter total alpha plus beta activity. Release of laundry waste at this concentration would result in a whole body exposure to a suitable sample of the population (as shown in the Commission Safety Analysis) of 0.033 mrem/year.

4.2.3 Other liquid wastes (including water from the high level solid waste burial ground) which contain or may be suspected to contain radioisotopes will be treated in the low level liquid waste treatment plant. The radioactivity in the feed to the treatment plant (via the interceptors and lagoon system) is limited to prevent excessive radiation levels near the lagoons and in the waste treatment plant, and to prevent overloading the low level waste treatment facility.

4.2.4 Except as specified in 4.2.2 all waste transferred to the lagoon system will be processed through the waste treatment plant to assure that as much of the radioactivity is removed as is

practicable with the installed equipment. For this same reason, all process steps in the waste treatment plant must be operating if the effluent is to be released from the lagoon system. During periods of process upsets or failures, the low level treatment plant effluent will be recycled.

4.2.5 The low level waste treatment plant is designed specifically to remove cesium and strontium from the aqueous wastes. Proper operation and control of the low-level liquid waste treatment plant, including continuous monitoring for Cs-137 at the exit of the ion-exchange unit, should assure that the concentrations of strontium and cesium are within specification limits at the time of discharge. Analyses for strontium-90 require time delays for the buildup of yttrium. Compliance with Specification 4.2.5 will be based on analytical results obtained after the waste has been released from the lagoon system.

4.2.6 Ruthenium and tritium are isotopes present in significant quantities in the waste transferred from the main plant and which will not be effectively removed by the waste treatment plant. Until additional treatment or other disposal is provided for these isotopes, they will be present in greater than MPC limits in the lagoon effluents and will be the predominate radioisotopes in Cattaraugus Creek. The con-concentrations of radioactive strontium and cesium in Cattaraugus Creek will be low; about 0.1% of the respective 10 CFR Part 20 limits. The action values given in Specification 4.2.6 are based on the recognition of the limitations of the present waste treatment system. The specification requires prompt action on the part of NFS to investigate releases above the predetermined values and to define and initiate corrective action.

4.14 EXHAUST FILTER DIFFERENTIAL PRESSURE

Applicability

This specification applies to the maximum differential pressures that may be maintained across the absolute filters in the exhaust ventilation systems.

Objective

To minimize the probability of filter failure in exhaust ventilation systems.

Specification

4.14.1 THE SUSTAINED OPERATING PRESSURE DIFFERENTIAL ACROSS ANY ABSOLUTE FILTER IN ANY OF THE EXHAUST VENTILATION SYSTEMS SHALL NOT EXCEED 85% OF THE GREATEST PRESSURE DIFFERENTIAL AT WHICH THAT FILTER HAS BEEN TESTED.

Basis

High Efficiency (absolute) filters are used in the plant's ventilation systems (Dissolver Off-Gas, Vessel Off-Gas, Main Building, Head End, Waste Tank Farm, DOG-VOG Backup and Low Level Waste Treatment Plant). The absolute filters act as the final containment of radioactivity prior to discharge to the plant stacks. The service of the absolute filters is limited by this specification to minimize the probability of filter failure. Specification 4.14.1 pertains to sustained service of the absolute filter; it is recognized that transient flow variations may cause differential pressure excursions exceeding the 85% limits as end of service life approaches.

The specifications for all absolute filters used at NFS include the common strength requirement that the absolute filter withstand a pressure drop of at least 10 inches W. G. without tearing or deforming. All absolute filters used at NFS are tested by the manufacturer to a pressure differential of at least 10 inches W. G.

5.1 EFFLUENT AND ENVIRONMENTAL MONITORING

Applicability

This specification applies to the sampling and analysis of gaseous and liquid plant effluents and to environmental monitoring.

Objective

To establish sampling points, sampling frequency and sample analytical requirements for gaseous and liquid plant effluents and to establish an environmental monitoring program and reporting requirement.

Specification

5.1.1 WHENEVER LIQUID IS DISCHARGED FROM THE STORAGE LAGOONS, A REPRESENTATIVE SAMPLE SHALL BE WITHDRAWN BY THE CATTARAUGUS CREEK CONTINUOUS SAMPLER. AT LEAST EVERY SEVEN DAYS, A COMPOSITE OF THIS SAMPLE SHALL BE ANALYZED FOR TRITIUM, RUTHENIUM 106, TOTAL ALPHA AND TOTAL BETA RADIOACTIVITY.

IF THE CONTINUOUS SAMPLER BECOMES INOPERATIVE, UP TO 2,000,000 GALLONS MAY BE DISCHARGED FROM THE LAGOONS IN ANY ONE-YEAR PERIOD PROVIDED CATTARAUGUS CREEK SAMPLES ARE COLLECTED NEAR THE CONTINUOUS SAMPLER LOCATION FOR EACH LAGOON DISCHARGE OR FOR EACH 100,000 GALLONS OF DISCHARGE, WHICHEVER RESULTS IN THE GREATER SAMPLING FREQUENCY. SAMPLES SO COLLECTED SHALL BE INDIVIDUALLY ANALYZED FOR TRITIUM, RUTHENIUM 106, TOTAL ALPHA AND TOTAL BETA RADIOACTIVITY.

5.1.2 THE RADIOACTIVITY IN STACK EFFLUENTS SHALL BE CONTINUOUSLY SAMPLED. THE SAMPLES SHALL BE ANALYZED AT LEAST EVERY SEVEN DAYS FOR IODINE 131 AND PARTICULATE RADIOACTIVITY. IF THE STACK SAMPLER BECOMES INOPERATIVE, IMMEDIATE REPAIR SHALL BE INSTITUTED TO RETURN THE SAMPLER TO SERVICE.

IN ADDITION TO THE WEEKLY SAMPLES OBTAINED FROM THE STACK, THE I-131, KR-85 AND PARTICULATE RADIOACTIVITY IN THE STACK GAS SHALL BE CONTINUOUSLY SUBJECT TO DETECTION BY A STACK MONITOR.

IF THE STACK SAMPLER BECOMES INOPERATIVE, IMMEDIATE REPAIR SHALL BE INSTITUTED TO RETURN THE UNIT TO SERVICE, AND THE STACK MONITOR SHALL BE USED IN THE INTERIM TO DETERMINE I-131 AND PARTICULATE RELEASES.

IF THE STACK MONITOR FAILS, A REPRESENTATIVE SAMPLE OF THE RADIOACTIVITY IN STACK EFFLUENTS SHALL BE COLLECTED EACH SHIFT AND USED TO DETERMINE I-131 AND DURING PERIODS AND PARTICULATE RADIOACTIVITY. THE Kr-85 CONTENT OF STACK GASES DURING PERIODS WHEN THE STACK MONITOR IS INOPERATIVE SHALL BE CALCULATED FROM FUEL BURNUP DATA.

5.1.3 THE FOLLOWING INFORMATION, DETERMINED FROM THE SAMPLES TAKEN AS REQUIRED BY 5.1.1 AND 5.1.2, (OR, FOR I-129, LAGOON SAMPLES) SHALL BE INCLUDED IN THE QUARTERLY OPERATING REPORT:

A) TOTAL CURIES OF ALPHA ACTIVITY DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

B) TOTAL CURIES OF BETA ACTIVITY DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

C) CURIES OF TRITIUM DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

D) CURIES OF RU-106 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

E) CURIES OF CS-137 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

F) CURIES OF CS-134 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

G) CURIES OF SR-89 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

H) CURIES OF SR-90 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

I) CURIES OF I-129 DISCHARGED INTO CATTARAUGUS CREEK EACH MONTH.

J) THE PERCENT OF 10 CFR 20.106a LIMITS FOR RADIOACTIVITY IN CATTARAUGUS CREEK AVERAGED FOR EACH RESPECTIVE MONTH OF THE REPORTING QUARTER.

K) THE PERCENT OF 10 CFR 20.106a LIMITS FOR RADIOACTIVE STRONTIUM AND CESIUM ISOTOPES IN CATTARAUGUS CREEK AVERAGED FOR EACH RESPECTIVE MONTH OF THE REPORTING QUARTER.

L) CURIES OF PARTICULATES DISCHARGED VIA THE STACK FOR EACH MONTH.

M) THE PERCENT OF THE MONTHLY TECHNICAL SPECIFICATION LIMIT FOR DISCHARGE OF PARTICULATES VIA THE STACK.

N) THE MAXIMUM PERCENT OF THE KR-85 DAILY TECHNICAL SPECIFICATION LIMIT IN THE STACK EFFLUENT FOR EACH MONTH.

O) THE CUMULATIVE PERCENT OF THE I-131 YEARLY TECHNICAL SPECIFICATION LIMIT IN THE STACK EFFLUENT REPORTED ON A MONTHLY BASIS.

Basis

Effluent Monitoring

Samples of water from Cattaraugus Creek are taken with a continuous sampler located about one-half mile downstream from the confluence of Cattaraugus and Buttermilk Creeks. Portions of each weekly sample are composited to product a sample representative of one month's discharge. The weekly samples are analyzed for gross alpha, gross beta, tritium and ruthenium 106. The composite is analyzed for gross alpha, gross beta strontium 90, strontium 89, cesium 134, cesium 137, ruthenium 106 and tritium.

A stream gage located near the sampling station is used to determine total flow in Cattaraugus Creek. Flow from the lagoon is determined by a calibrated weir located on the discharge line. The flow ratio is used in conjunction with the iodine 129 analysis of lagoon water to calculate the iodine 129 content in Cattaraugus Creek.

Gaseous plant effluents are sampled in the plant stack. The stack sampler contains a filter to collect particulates and an activated charcoal filter to collect iodine 131. Samples are removed from the stack sampler at least each week and analyzed.

The stack monitor is used to continuously determine the particulate, iodine 131 and krypton 85 radioactivity in the stack air and to alert operators if pre-set limits are exceeded. The particulate sampler is changed every 8 hours and will alarm if the accumulation of particulate radioactivity over an 8-hour period exceeds that which

would occur if particulates were being discharged over the same time period at the limit set by Technical Specification 4.1. The iodine 131 monitor, a carbon filler paper absorber, is changed on a weekly cycle. The instrument is set to alarm if, over a week's period, the counting rate increases 10,000 counts/minute. This is far below the increase in count rate (about 400,000 counts/minute) which would occur if iodine were released for one week at a rate of 1/52 of the yearly technical specification limit. The krypton 85 monitor is set to alarm if the counting rate exceeds five times the background rate. The alarm point is not, in this case, directly related to allowable release rates. The instrument will alarm only during periods of high krypton 85 discharge, for example during dissolution of high burn-up fuels.

5.1.4 THE LICENSEE SHALL CONDUCT THE ENVIRONMENTAL MONITORING PROGRAM GIVEN IN TABLE 5.1 AND SHALL REPORT THE RESULTS OBTAINED AT THE END OF EACH SIX-MONTH PERIOD BEGINNING FROM JANUARY 1, 1971. SUCH REPORTS WILL BE DUE WITHIN 60 DAYS AFTER THE END OF EACH REPORTING PERIOD. IF QUANTITIES OF RADIOACTIVE MATERIAL RELEASED DURING THE REPORTING PERIOD ARE UNUSUAL FOR NORMAL PLANT OPERATIONS, INCLUDING EXPECTED OPERATIONAL OCCURRENCES, THE REPORT SHALL COVER THIS SPECIFICALLY.

TABLE 5.1

EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

A. The licensee shall determine for each month, based on samples taken at the indicated location and other information collected during the month, the following:

i	Total curies of beta radioactivity other than tritium released at effluent weir.
ii	Total curies of alpha radioactivity released at the effluent weir.
iii	Curies of tritium released at effluent weir.
iv	Curies of strontium 90 released at effluent weir.
v	Curies of cesium 134 released at effluent weir.
vi	Curies of cesium 137 released at effluent weir.
vii	Curies of ruthenium 106 and rhodium 106 released at effluent weir.
viii	Volume of water released at effluent weir.
ix	Volume of water flow through site in Cattaraugus Creek.
x	Curies of beta emitting particulates released via the stack.
xi	Curies of alpha emitting particulates released via the stack.
xii	Curies of iodine 131 released via the stack.
xiii	Curies per cubic meter gross beta radioactivity of particulates (average and maximum) collected by each of the 3 site perimeter samplers on filter paper.
xiv	Curies per cubic meter gross alpha radioactivity of particulates (average and maximum) collected by each of the 3 site perimeter samplers on filter paper.

B. The licensee shall determine each quarter, based on samples and other information collected during the quarter, the following:

i Identity of principal radionuclides whose presence can be determined by gamma spectroscopy of a sample (one per calendar quarter) of Buttermilk Creek bottom silt collected at the Thomas Corner Road bridge.

ii Microcuries per gram total beta radioactivity of a sample (one per calendar quarter) of Buttermilk Creek bottom silt collected at the Thomas Corner Road bridge.

iii Microcuries per gram total alpha radioactivity of a sample (one per calendar quarter) of Buttermilk Creek bottom silt collected at the Thomas Corner Road bridge.

iv Microcuries tritium per milliliter of water collected once per quarter at the Buttermilk Creek silt sampling location during a normal liquid effluent release from the plant.

v Microcuries total beta radioactivity per milliliter of water collected once per quarter at the Buttermilk Creek silt sampling location during a normal liquid effluent release from the plant.

vi Microcuries total alpha radioactivity per milliliter of water collected once per quarter at the Buttermilk Creek silt sampling location during a normal effluent release from the plant.

vii Curies of strontium 89 released at the effluent weir based on an analysis of a composite of the liquid released during the last month of the quarter.

viii Curies of zirconium 95 and niobium 95 released at the effluent weir based on an analysis of a composite of the liquid released during the last month of the quarter.

ix Curies of iodine 129 released at effluent weir.

x A quantitative analysis of alpha emitting components in a quarterly composite collected at the effluent weir.

(Change No. 15)

xi The exposure, in megawatts days, of fuel dissolved during the quarter.

xii The curies of krypton 85 released via the stack as determined using the stack monitor.

xiii Curies of tritium released via the stack.

xiv Curies of iodine 129 released via the stack.

xv Curies of strontium 89 released via the stack.

xvi Curies of strontium 90 released via the stack.

xvii Curies of ruthenium 106 released via the stack.

xviii Curies of cesium 134 released via the stack.

xix Curies of cesium 137 released via the stack.

xx Microcuries per milliliter tritium (average) in precipitation collected at each of the 3 site perimeter sampling stations.

xxi Milliroentgen of radiation (including background) accrued each quarter at each of the 3 site perimeter sampling stations.

xxii The maximum value at ground level offsite in each $\pi/8$ sector about the stack of the X/Q integrated for each seasonal quarter from hourly averages of continuously recorded meteorological data. The X/Q shall be determined using equation 3.144 given on page 113 of Meteorology and Atomic Energy (1968). Seasonal quarters are defined as:

Spring	March through May
Summer	June through August
Fall	September through November
Winter	December through February

xxiii The curie seconds per cubic meter maximum exposure of Kr-85 integrated for the reported quarter at ground level offsite in each $\pi/8$ sector about the stack, computed from meteorological data recorded during emission of the respective increments of Kr-85 that register above background on the Kr-85 stack gas monitor.

C. The licensee shall determine the following information according to the schedule given:

i If, during any week from April to October, inclusive, the release of iodine 131 from the stack exceeds 30 millicuries, the licensee shall determine the microcuries per milliliter of iodine 131 in samples of milk from farms within 2-1/2 miles of the plant in each direction expected to receive the release. One sample shall be collected each week for the four weeks following the release. Each sample shall be a composite of one day's production from cows which are on pasture.

ii During August of each year the licensee shall collect a milk sample from a farm within 2-1/2 miles of the plant in the north-west sector and a milk sample from a farm within 2-1/2 miles of the plant in the north-east sector. The samples shall be composites of one day's production from cows which are on pasture. Each sample shall be analyzed separately to determine:

a) Microcuries iodine 129 per milliliter

b) Microcuries strontium 90 per milliliter

c) Microcuries cesium 134 per milliliter

d) Microcuries cesium 137 per milliliter

iii During the second and third quarters of each year, licensee shall take fish samples from Cattaraugus Creek between the Springville hydroelectric dam and the Cattaraugus Creek-Buttermilk Creek confluence. One sample shall be taken during each of the two quarters. A sample shall consist of at least 9 fish, each at least six inches long. Each fish in each sample shall be analyzed to determine:

a) Median and geometric deviation of cesium 134 microcuries per kilogram of flesh.

b) Median and geometric deviation of cesium 137 microcuries per kilogram of flesh.

c) Median and geometric deviation of strontium 90 microcuries per kilogram of flesh.

d) Median and geometric deviation of strontium 90 microcuries per kilogram of skeleton.

iv Once per year the licensee shall determine the following from a deer taken onsite during the hunting season and taken, preferably, north of the plant:

a) Microcuries cesium 134 per kilogram of flesh.

b) Microcuries cesium 137 per kilogram of flesh.

c) Microcuries strontium 90 per kilogram of flesh.

d) Microcuries strontium 89 per kilogram of skeleton.

e) Microcuries strontium 90 per kilogram of skeleton.

D. The environmental reports to be prepared by the licensee shall contain descriptive material covering the environmental surveys performed during the reporting period including information on:

i The number and types of samples taken; e.g., air, surface water, soil, fish.

ii The number and types of measurements made.

iii Locations of the sample points and monitoring stations.

iv The frequency of the surveys.

v A summary of survey results.

E. If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 3% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based should be provided.

F. If statistically significant variations in offsite environmental concentrations with time are observed, correlation of these results with effluent releases should be provided.

Basis

Environmental Monitoring

The environmental survey incorporates measurements to provide background data and measure possible plant effects. Samples collected at points where concentrations of effluents in the environment are expected to be greatest will be compared with samples collected concurrently at points expected to be essentially unaffected by station effluents. The latter samples will provide background measurements as a basis for distinguishing significant radioactivity introduced into the environment by the operation of the plant from that due to other sources such as nuclear detonations.

This schedule will assure that significant changes in the environmental radioactivity are sampled most frequently. Those which are less affected by transient changes but show long-term accumulations are sampled less frequently.

The monitoring program will be reviewed and re-evaluated periodically after data such as composition, quantity and frequency of releases, dilution factors and measured concentrations in food and other organisms (if any are observed) are obtained.

6.5 AIR FILTRATION EQUIPMENT

Applicability

This specification applies to 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 ROUTINE SERVICE, WITH PARTICULATES OF AN APPROPRIATE SIZE TO ESTABLISH THAT THE INSTALLED FILTERS PROVIDE A COLLECTION EFFICIENCY OF AT LEAST 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
MAIN BUILDING	15T-49, 15T-49A
DOG-VOG BACKUP	6T-3, 6T-3A
HEAD END	15T-72, 15T-72A, 15T-73, 15T-73A
LOW LEVEL WASTE TREATMENT PLANT	479-401-B, 479-402-B

6.5.2 THE DIFFERENTIAL PRESSURE ACROSS THE FINAL OPERATING ABSOLUTE FILTERS (OR ABSOLUTE PLUS ROUGHING WHERE THE FILTERS ARE WITHIN THE SAME FRAME) SHALL BE RECORDED BY INSTRUMENTATION.

6.5.3 ALARMS SHALL BE PROVIDED TO ANNUNCIATE HIGH DIFFERENTIAL PRESSURE ON THE FINAL OPERATING ABSOLUTE FILTERS OF THE MAIN BUILDING, HEAD END, WASTE TANK FARM AND DOG-VOG BACKUP SYSTEMS. THESE ALARMS SHALL BE SET TO ANNUNCIATE AT LESS THAN 75% OF THE GREATEST DIFFERENTIAL PRESSURE AT WHICH THAT FILTER HAS BEEN TESTED.

6.5.4 AN ALARM SHALL BE PROVIDED TO ANNUNICATE LOW DIFFERENTIAL PRESSURE ON THE FINAL OPERATING ABSOLUTE FILTER IN THE MAIN BUILDING.

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 HANDLING ABNORMAL OPERATING CONDITIONS AND FOR RADIATION PROTECTION SHALL BE PROVIDED.

7.1.5 LOGS AND RECORDS SHALL BE MAINTAINED AS REQUIRED BY THE PROVISIONAL LICENSE AND SHALL INCLUDE MINUTES OF MEETINGS OF THE PLANT SAFETY COMMITTEE AS WELL AS DETAILS OF INVESTIGATIONS AND RECOMMENDATIONS IN REGARD TO ABNORMAL OCCURRENCES AND ACTIONS TAKEN THEREON.

7.1.6 AUDIT OF OPERATIONS. MANAGEMENT SHALL PROVIDE FOR PERIODIC AUDITS OF THE OPERATIONS THROUGHOUT THE ENTIRE FACILITY WITH REGARD TO RADIOLOGICAL AND NUCLEAR SAFETY. THESE AUDITS SHALL BE PERFORMED BY COMPETENT COMPANY OR OTHER TECHNICAL PERSONNEL NOT DIRECTLY RESPONSIBLE FOR THE OPERATION AUDITED.

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

7.1.7.1 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.

7.1.7.2 IMMEDIATE STEPS SHALL BE TAKEN TO ESTABLISH THE CAUSE OF THE OCCURRENCE IN 7.1.7.1 BY RESPONSIBLE MANAGEMENT REVIEW.

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

Safety Evaluation

By

The Division of Materials Licensing

Nuclear Fuel Services, Inc.

Docket No. 50-201

Low Level Waste Treatment Plant

Authorization Requested

By letter dated January 14, 1971, supplemented by additional information submitted on April 2, 1971, Nuclear Fuel Services, Inc. (NFS) submitted their description and safety analysis relevant to the design and construction of new facilities for the reduction of radioactivity contained in low level liquid wastes and requested authorization for operation of the new facilities. Authorization for operation of the Low Level Waste Treatment Plant has been granted by changes in Technical Specifications 4.2 Radioactivity Content of Released Liquid Effluents, 4.14 Exhaust Filter Differential Pressure, 5.1 Effluent and Environmental Monitoring, 6.5 Air Filtration Equipment and 7.1 Administrative Requirements.

Background

Since the plant began operating in April 1966, the concentrations of radioactive materials in Cattaraugus Creek resulting from discharges of low level liquid wastes from the site have been higher than were projected in the NFS Final Safety Analysis Report. In a letter to NFS dated May 31, 1968, the Commission stated that releases from the NFS plant should be significantly reduced. In reply, NFS, by application dated July 9, 1969, requested authorization for deep-well disposal of low-level liquid waste. Questions related to public health and safety raised by the deep-well disposal concept could not be resolved at that time; therefore, the Commission informed NFS by a letter dated May 27, 1970, that NFS should develop an alternate plan to achieve a reduction in quantities and concentrations of radioactivity released in liquid effluents. NFS proposed to use a chemical treatment and ion exchange system for reduction of radioactivity in effluents (letter dated June 29, 1970). The Commission accepted the proposed treatment plan as a first-step toward meeting the overall objective which is ultimately to maintain radioactivity in liquids at the

point of release from the lagoon at or below the levels set forth in Appendix B, Table II of 10 CFR Part 20 (letter dated August 8, 1970). In accepting this first step the observation was made in the Commission letter that it "appears to be feasible and compatible for use with follow-on systems for meeting the stated objective, but we believe every effort should be made to meet the objective stated in our May 27 letter, at least for cesium and strontium within the next six to nine months." On May 4, 1971 permission was given to NFS for test operation of the low level waste treatment plant.

Description

The Low Level Waste Treatment Plant consists of flocculation, clarification and ion exchange steps. In the flocculation step, sodium hydroxide, ferrous sulfate and clay are added to the waste water feed to the treatment plant to precipitate most of the calcium and magnesium present as hydroxides and to form a gelatinous precipitate of ferric hydroxide. Cesium, strontium and other radioisotopes are absorbed on the precipitate and on the clay added to the feed waste water.

The precipitated and absorbed materials are removed from the water in a sludge-blanket clarifier. Feed to the clarifier is introduced through a central well. Purified water overflows from the top of the clarifier after passing upward through the sludge blanket. Settled sludge is discharged continuously from the bottom cone of the clarifier, centrifuged and packaged in drums for burial on site.

Final traces of suspended solids are removed from the clarifier overflow by pumping the water through a fixed-bed anthracite coal filter. The water is then passed through a fixed-bed ion exchange unit containing a cesium-selective ion exchange resin (such as Duolite ARC-359) in the sodium form. The pH of the effluent from the ion exchange unit is adjusted with sulfuric acid to meet New York State water quality limits as it flows to the lagoon system before being released to the water shed.

Backwash water from the anthracite filter, ion exchange regeneration solutions and the centrifuge supernatant waste are recycled to the feed of the treatment plant.

Overall operation of the treatment plant is monitored by the use of instruments equipped with alarms in the main plant control room. The ion exchange effluent is continuously monitored with an in-line instrument set to detect gamma radiation from cesium isotopes.

Overall decontamination factors, as reported by NFS, are expected to be about 2900 for strontium, 250 for cesium and 3.6 for ruthenium.

The new facility is located in a separate structure north of existing Lagoon No. 2. Appropriate consideration has been given in the design of the facility to protection of plant operating personnel from direct radiation or contamination by isolation and shielding of equipment and by ventilation systems. Working times, in the facility, as stated by NFS, will be limited; the unit is expected to require a minimum of attention.

Safety Considerations

1. General

The Low Level Waste Treatment Plant presents no new or unusual problems for protection of plant operating personnel from radiation exposure. Existing plant procedures and practices will be followed for radiological protection of personnel. The inventory of activity which could be present in process equipment is estimated by NFS to be about 40 curies of beta activity. Our independent evaluation of the release of the activity under tornado conditions (and assuming that all the activity is strontium 90) indicates that the maximum ground level concentration would be equivalent to a 50 day exposure at the concentration limit for strontium 90 as given in 10 CFR 20, Appendix B, Table II for unrestricted areas. The radioactivity in the plant will be primarily in the form of insoluble materials and no off-site dispersion would occur under earthquake conditions. Based on these considerations, we do not consider it necessary to build the waste treatment plant to resist tornados or earthquakes.

NFS estimates the gross alpha content of the feed to the plant to be 1.2×10^{-6} microcuries per milliliter. Because of the low concentration of total plutonium and uranium in the feed to the waste treatment plant, there will be no criticality hazard.

2. Effluent Discharges

The radioactivity which was present in Cattaraugus Creek due to effluent discharged from the NFS plant from 1967 through 1970 is shown in Table I. The yearly average percentage of the 10 CFR Part 20 limits in Cattaraugus

Creek is also given in the table.[1] More complete isotopic analyses are not routinely made, but, some studies have indicated that the gross beta activity (exclusive of tritium) is composed of about 75% ruthenium 106, 7% strontium 90, 6% cesium 134 and 137 with the remainder composed of other isotopes none of which exceeds 1% of the total beta activity. Because of its low individual 10 CFR Part 20 limit (3×10^{-7} μc/ml), strontium 90 accounts for about 75% of the reported percentage of the 10 CFR Part 20 limit in Cattaraugus Creek. Cesium isotopes are the next most important contributors to the yearly average values.

The new waste treatment plant is specifically designed to remove strontium and cesium isotopes from the plant effluents. These isotopes are the most limiting because they can reconcentrate in foods eaten by man.

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(1) The percentage of the 10 CFR Part 20 limits is calculated using the following formula:

$$\left[\frac{C_A}{MPC_A} + \frac{C_B}{MPC_B} + \ldots \frac{C_1}{MPC_1} \right] \times 100 = \%\ 10\ \text{CFR}\ 20\ \text{Limit}$$

Where C_A, C_B, C_1 is the concentration in μc/ml of radioisotope A, B or 1 in Cattaraugus Creek.

MPC_A, MPC_B, MPC_1 is the 10 CFR Part 20 limit for radioisotope A, B or 1 in water in unrestricted areas.

MPC for gross beta activity, if a separate analysis has been made for Sr-90, is taken as 1×10^{-5} μc/ml, the value given in 10 CFR 20 Appendix B, Table II, for soluble ruthenium.

MPC for gross alpha activity is taken as 5×10^{-6} μc/ml, the value given in 10 CFR 20, Appendix B, Table II for soluble plutonium.

TABLE I

RADIOACTIVITY IN CATTARAUGUS CREEK

1967-1970

Year	Curies Gross α	Gross β	H-3	Sr-90	Yearly Average Concentration in Cattaraugus Creek as % of Limit
1970	0.105	87.0	4516	14.2	21.9
1969	0.376	136.2	5975	10.07	19.7
1968	0.139	46.7	2662	5.09	9.1
1967	0.056	38.7	4204	9.09 (1)	15.51

(1) For period March through December.

The effects of future discharges of strontium and cesium isotopes from the NFS plant on radioactivity in Cattaraugus Creek can be estimated based on technical specification limits and the capacity of the low level treatment plant. The action value for releases of strontium and cesium isotopes, as given in specification 4.2.5, is defined by the following equation:

$$\frac{C_{Cs-134}}{9 \times 10^{-6}} + \frac{C_{Cs-137}}{2 \times 10^{-5}} + \frac{C_{Sr-89}}{3 \times 10^{-6}} + \frac{C_{Sr-90}}{3 \times 10^{-7}} = 1$$

(Where C = concentration, in microcuries per milliliter, of the indicated isotope in the treated waste)

Assuming that the effluent discharge is 200,000 gallons per day (the capacity of the waste treatment plant) for a full year, at maximum allowable concentrations and assuming that each isotope contributes equally to the total action value, i.e. that the ratio of the concentration of each radioisotope to its Part 20 limit is 0.25, the total releases would be 0.63 curies of cesium 134, 1.38 curies of cesium 137, 0.2 curies of strontium 89 and 0.02 curies of strontium 90. At these release levels the concentration of cesium and strontium radioisotopes in Cattaraugus Creek on an annual average basis would be about 0.1% of 10 CFR Part 20 limits as compared to about 15% of 10 CFR Part 20 limits in 1969 and 1970.

Tritium will not be removed by the new waste treatment plant and ruthenium removal will be low. Releases of these isotopes, which in the future will be the predominant radioisotopes in Cattaraugus Creek, are controlled by Technical Specification 4.2.6 which sets an action value of 10% of 10 CFR Part 20 limits for all radioisotopes in Cattaraugus Creek. Although it is highly unlikely to occur, an upper level for release of tritium or ruthenium radioisotopes can be projected by assuming that the concentration of the individual isotope in Cattaraugus Creek, on an annual average basis, is at the 10% action value. Based on these assumptions the projected release would amount to 95,000 curies of tritium or 320 curies of ruthenium 106 released to Cattaraugus Creek each year. Actual releases will be below the maximum values listed because both isotopes will be present and the technical specification limit applies to all radioisotopes present. Realistic estimates of future actual tritium and ruthenium releases cannot be made at present because of uncertainties such as (1) the fate of tritium in reactor fuels (particularly Zircaloy clad fuels,

(2) the ruthenium decontamination factor in the waste treatment plant (3) future plant processing capacity, (4) burn-up of future fuels and (5) the effectiveness of future plant modifications to be made to improve ruthenium removal.

The significance of the projected future releases of radioisotopes in liquid effluents from the NFS plant, in terms of population exposures, is discussed in the following sections of this analysis.

3. Population Exposure Estimate

(a) Basis for Estimates

Population exposures from discharges of liquid effluents from the NFS site result primarily from ingestion of fish and venison. The Public Health Service of the U.S. Department of Health, Education and Welfare has studied population exposure around the NFS site[1] and effluent discharges from the site[2], and the results of these studies can be used as a basis for estimating future population exposures.

The steps involved in establishing a basis for estimating future population exposure are:

1) Establish the relationship between consumption of specified quantities of fish and venison containing known quantities of radioactivity and the personnel exposures received.

2) Establish the relationship between the amount of radioactivity in fish and venison and the quantity of activity discharged from the site.

3) Combining 1) and 2) and relating exposures through ingestion of fish and venison to the activity in plant effluents and expressing the results as the organ dose per curie discharged for each important radioisotope.

Estimates made by the Public Health Service of the intake of radioisotopes by individuals in the population near NFS in 1968 by eating fish and venison and the resulting 50 year dose commitment to members of that population are shown in Table II.

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(1) BRH/NERHL 70-1, "An Estimate of Radiation Doses Received by Individuals Living in the Vicinity of a Nuclear Fuel Reprocessing Plant in 1968."

(2) BRH/NERHL 70-2, "Liquid Waste Effluents from a Nuclear Fuel Reprocessing Plant."

Table II shows that the 50 year dose commitment of an average individual in the population group eating fish and venison in 1968 is about 2.2 mrem to the whole body from cesium 134, cesium 137 and tritium; 17.6 mrem to bone from strontium 90; and 0.96 mrem to the G. I. track from ruthenium 106 and cobalt 60. We estimate the population exposed through consumption of fish and venison to be between 100 and 1,000 people.

In order to relate population exposure in 1968 to releases of activity from the NFS site, it is necessary to estimate the 1968 curie discharges of specific isotopes. In 1968, NFS reported that liquid effluents contained 2,662 curies of tritium, 5.09 curies of strontium 90 and 41.6 curies of other beta activity. More detailed isotopic analysis of 1968 plant effluents are not available. In 1969, the Public Health Service evaluated the isotopic content of NFS liquid effluents in greater detail during the period May through October. Assuming that the isotopic composition (exclusive of tritium and strontium 90 for which separate analyses are available) of the effluent in 1968 is the same as in 1969, the 41.6 curies of unidentified beta activity released in 1968 would contain 1.12 curies of cesium 134, 4.56 curies of cesium 137, 29.6 curies of ruthenium 106, and 0.112 curies of cobalt 60. Using these assumed releases, in conjunction with Table II exposures, the maximum organ dose per curie discharged for each important radioisotope can be estimated. Results of these calculations are given in Table III.

(b) Future Maximum Exposure Estimates

The organ dose per curie discharged. derived as shown in Table III, can be used in conjunction with estimates of future maximum releases of radioisotopes to make an upper level estimate of future population exposures.

In Section 2 of this analysis, the following estimates for future releases of important radioisotopes were given.

Isotope	Curies/Year
Sr-90	0.02
H-3	95,000
Cs-134	0.63
Cs-137	1.38
Ru-106	320

TABLE II

1968 Exposure Estimates for Persons Eating Fish and Venison

Nuclide	Annual Average Intake: Fish, 5.5 kg/yr[3] µCi/kg[2]	Fish, 5.5 kg/yr[3] µCi	Venison, 2. kg/yr[3] µCi/kg[2]	Venison, 2. kg/yr[3] µCi	µCi/yr From Fish + Venison	Dose Commitment Factor (mrem per 50 years per µCi intake)	50 Year Dose Commitment mrem	Organ
Sr-90	3.4×10^{-4}	1.87×10^{-3}	1.1×10^{-4}	2.2×10^{-4}	2.09×10^{-3}	8400.[5]	17.6[5]	Bone
H-3	1.2×10^{-2}	6.6×10^{-2}	3×10^{-2}	6×10^{-2}	1.26×10^{-1}	.13	.016	Whole Body
Cs-134	--------	--------	2.6×10^{-3}	5.3×10^{-3}	5.2×10^{-3}	80	.416	Whole Body
Cs-137	6.3×10^{-4}	3.47×10^{-3}	1.3×10^{-2}	2.6×10^{-2}	2.95×10^{-2}	60	1.768	Whole Body
Ru-106	8.5×10^{-4}	4.68×10^{-3}	--------	--------	4.68×10^{-3}	195	.912	G.I. tract
Co-60	2.1×10^{-4}	1.16×10^{-3}	--------	--------	1.68×10^{-3}	40	.046	G.I. tract

1. Based on BRH/NERHL 70-1 data.
2. Ibid. p. 8, Table 3.
3. Ibid. p. 6, Table 2.
4. Ibid. p. 5, Table 1.
5. The first year dose for Sr-90 ingestion is about 7% of the 50 year dose commitment or 1.2 mrem. For other radionuclides listed, the first year dose is >90% of the 50 year dose commitment.

TABLE III

ORGAN DOSE PER CURIE DISCHARGED FOR PERSONS EATING

FISH AND VENISON

Nuclide	Curies Discharged in 1968	1968 50 Year Dose Commitment mrem	50 Year Dose Commitment mrem per Ci Discharged	Organ
Sr-90	5.09	17.5	3.43	Bone
H-3	2,662	.016	6×10^{-6}	Whole Body
Cs-134	1.12	.416	.37	Whole Body
CS-137	4.56	1.768	.39	Whole Body
Ru-106	29.6	.912	.03	G.I. tract
Co-60	.112	.046	.41	G.I. tract

The curie discharge estimates for strontium and cesium radioisotopes were based on discharge of a mixture of radioisotopes. If only individual radioisotopes were present, the curie discharge limit for each radioisotope could increase by a factor of four without exceeding technical specification 4.2.5. The following estimates of the upper level of future population exposures were based on releases of individual radioisotopes even though it was recognized that resulting exposures would be overstated.

In Table IV the future maximum discharges of individual radioisotopes and the corresponding upper limits of future population exposures are given. Maximum first year bone and whole body exposures are less than 2% of the FRC guidance limits for a suitable sample of the population. There is no specific FRC guidance for G. I. tract exposure; however, the calculated exposure of 9.6 mrem/year is low compared to the FRC whole body guidance limit of 170 mrem per year.

The exposures estimated above are overstated. Future population exposures will actually be substantially lower for the following reasons:

1. The above estimates are based on releases of isotopes at maximum individual release levels. In actual plant operation, mixtures of isotopes will be present and no individual isotope will be at its limit.

2. The estimates are based on discharges over a full year period at maximum permissible concentrations and at maximum volume flows, and

3. Actual releases of ruthenium will be lower than those estimated because the waste treatment plant is expected to have a ruthenium decontamination factor of about four and future plant modifications are planned which should improve in-plant ruthenium retention.

Radioisotopes other than tritium, ruthenium, cesium and strontium which are present in plant wastes are not expected to add significantly to population exposure. No radioisotope other than tritium, ruthenium, cesium and strontium will occur in plant effluent at the

TABLE IV
MAXIMUM ESTIMATED
AVERAGE EXPOSURE LIMIT OF
PERSONS EATING FISH & VENISON

Nuclide	Fraction of Appendix B Table II, Col. 2 Concentration limit	Possible Ci/yr Discharge	50 Year Dose Commitment per Year of Plant Operation mrem	Organ
Sr-90	1.0 (1)	0.08	0.3 (3)(4)	Bone
H-3	0.1 (2)	95,000	0.57	Whole Body
Cs-134	1.0 (1)	2.49	0.92	Whole Body
Cs-137	1.0 (1)	5.52	2.15	Whole Body
Ru-106	0.1 (2)	320.	9.6	G.I. Tract

(1) At point of discharge from lagoon system.

(2) In Cattaraugus Creek

(3) The dose accrued in the first year for Sr-90 is about 0.02 millirem. First year dose accruals for other radioisotopes and >90% of the 50 year dose commitment.

(4) Other dose commitment models could be used. For example, a dose commitment of 2000 mrem/μcc intake can be derived by relating continuous consumption of water containing the maximum permissible concentration of Sr-90 (3×10^{-7} μc/ml) to the resulting FRC estimated exposure of 500 mrem/year to bone. On this basis, the 50 year dose commitment per year of plant operation for Sr-90 would be less than 0.1 mrem.

TABLE V

LIMIT OF FISH & VENISON EATING GROUP EXPOSURE CONTRIBUTABLE BY UNTREATED WASTE STREAMS

	Steam Condensate	Laundry Effluent
Annual Flow, ml(1)	3.2×10^{11}	6.9×10^{9}
Gross Activity Limit, μc/ml	3×10^{-7}	1×10^{-4}
Gross Release Limit, Ci/yr	.096	.69
Ratio of Release limit to 1968 release.	.002	.015
Bone Exposure, mrem/yr	.0025	.018
Whole Body Exposure, mrem/yr	.0044	.033
G.I. Tract Exposure, mrem/yr	.0019	.014

(1) Based on boiler capacity of 80,000 pounds per hour and no condensate recycle and the NFS reported value for laundry wastes of 19,000 liters per day average.

lagoon outfall above their individual 10 CFR Part 20 limits. The total concentration of all radioisotopes, exclusive of tritium, ruthenium, cesium and strontium, after dilution in Cattaraugus Creek and assuming no removal in the treatment plant should not exceed about 0.1% of 10 CFR Part 20 limits.

The effect of two other streams discharged to the environment should be considered. These are the steam condensate and laundry discharges. Steam condensate, according to specification 4.2.1, may be discharged to the environment if it has been determined that the gross alpha plus beta activity (exclusive of tritium) is less than 3×10^{-7} microcuries per milliliter. Laundry waste (specification 4.2.2) may be transferred to the lagoon system downstream of the waste treatment plant if the gross alpha plus beta activity is less than 1×10^{-4} microcuries per milliliter. The total curie releases in these streams and exposures calculated according to the organ dose per curie release concept described above is shown in Table V which shows that discharge of these effluents have negligible effects on population exposure.

Conclusion

Based on our review of the Low Level Waste Treatment Plant design and operation, we conclude that it does not present significant hazard considerations not described or implicit in the NFS "Final Safety Analysis Report" and that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the new facility. We further conclude that cesium and strontium radioactivity in liquid effluents discharged from the site will be substantially reduced when the waste treatment plant is placed in operation.

Approval of the attached Technical Specifications relating to operation of the Low Level Waste Treatment Plant is recommended.

Signed:

W. A. Nixon

W. A. Nixon
Irradiated Fuels Branch
Division of Materials Licensing

Approved:

R. B. Chitwood, Chief
Irradiated Fuels Branch
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