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MASTER

SAFETY EVALUATION

BY THE

DIVISION OF MATERIALS LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ALLIED-GULF NUCLEAR SERVICES

ALLIED CHEMICAL NUCLEAR PRODUCTS, INC. AND

GULF ENERGY & ENVIRONMENTAL SYSTEMS, INC.

BARNWELL NUCLEAR FUEL PLANT

DOCKET NO. 50-332

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1.0 INTRODUCTION

Allied-Gulf Nuclear Services, a partnership, and Allied Chemical Nuclear Products, Inc., (a wholly-owned subsidiary of Allied Chemical Corporation) and Gulf Energy and Environmental Systems, Inc., (a wholly-owned subsidiary of Gulf Oil Corporation) [hereafter collectively referred to as "Allied-Gulf"] filed with the Atomic Energy Commission (AEC) an application dated November 6, 1968, with amendments (designated by the applicant as Amendments Nos. 1, 2, 3, 4, and 5, and Addenda Nos. 1, 2, 3, 4, 5, and 6) for all necessary licenses to construct and operate an irradiated nuclear fuel recovery plant, the Barnwell Nuclear Fuel Plant, pursuant to Section 104 b. of the Atomic Energy Act of 1954, as amended. Allied-Gulf proposes to build the Barnwell Nuclear Fuel Plant (BNFP) on a site about seven miles west of the town of Barnwell, in Barnwell County, South Carolina. The site is contiguous with the eastern boundary of the AEC Savannah River Plant (SRP).

The BNFP will process irradiated nuclear power reactor fuel consisting of uranium oxide, or a mixture of plutonium oxide and uranium oxide, clad in stainless steel or zirconium alloys. The plant is designed to process 1,500 metric tons of uranium (MTU) per year at an average rate of 5 MTU/day. An average fuel process batch will consist of irradiated reactor fuel elements that will have contained up to about 3.5 percent U-235, or 29 kg fissile plutonium per MTU, prior to irradiation. The average burnup for a given fuel process batch will be

less than 40,000 megawatt-days per ton (MWd/MTU) at 50 MW/MTU. However, from time to time a fuel process batch with greater fissile content (up to 5 percent U-235 or equivelant Pu) or with higher fission product content (higher burnup or power density) may be processed by appropriate reduction in the plant's throughput rate. Prior to processing, the reactor fuel will be aged at least 90 days and normally will be aged about 160 days from reactor discharge.

This Safety Evaluation discusses the principal plant features and presents our evaluation of safety considerations related to the Barnwell Nuclear Fuel Plant. In performing our review, we considered first the Barnwell site regarding its suitability in relation to the possible effluents that might be released under normal, transient and accident conditions at the BNFP. Second, we considered the confinement components and systems. Critical structures were evaluated according to criteria governing natural phenomena design bases for nuclear reactors. Third, we evaluated the Purex process system to be used in the BNFP with respect to whether margins of safety are adequate to mitigate transient and possible upper limit accident conditions. Fourth, we evaluated the accident prevention and protection features in the BNFP. Finally, we evaluated the BNFP as a whole with respect to its primary safety function, i.e., radiological protection to persons in the plant and in the public. Future plant operations, engineered safety features and potential accidents were evaluated based upon the plant operating at its design capabilities.

Our safety evaluation is based upon Allied-Gulf's amended Preliminary

Safety Analysis Report (PSAR). In the course of the review of the material submitted, we held a number of meetings with representatives of Allied-Gulf to discuss the proposed plant. As a consequence, we requested additional information which was provided in several amendments and addenda. A chronology of our review is attached as Appendix H.

Our technical evaluation of the preliminary design of the proposed plant was accomplished with the assistance of consultants. Appendices B through E include the reports of our consultants on meteorology, geology and hydrology, seismology, environmental considerations, and structural design.

The Commission's Advisory Committee on Reactor Safeguards (ACRS) has also conducted a review of the application and has met with both Allied-Gulf and the staff. A copy of its report to the Commission on the Barnwell Nuclear Fuel Plant is included as Appendix A.

Based on evaluations of the information provided in the PSAR by Allied-Gulf, both we and the ACRS have concluded that there is reasonable assurance that the proposed BNFP can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

The application and amendments thereto are available for public inspection at the AEC's Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the Barnwell County Courthouse, Barnwell, South Carolina.

The construction permit review of this facility is the first

stage of our continuing review of the design, construction, and operating features of the BNFP. Prior to issuance of an operating license, we will review the final design to determine that all of the Commission's safety requirements have been met. The BNFP would then be operated only in accordance with the terms of the operating license and the Commission's regulations under our continued surveillance.

The issues to be considered, and on which findings must be made by an Atomic Safety and Licensing Board before the requested construction permit may be issued, are set forth in the Notice of Hearing published in the Federal Register, 35 F.R. 14170, September 5, 1970.

2.0 FACILITY DESCRIPTION

The function of the Barnwell Nuclear Fuel Plant (BNFP) is to recover both uranium and plutonium from irradiated fuel elements which are discharged from light-water moderated and cooled nuclear power reactors. The plant will be capable of processing 1500 metric tons of uranium annually, which was contained in fuel elements that prior to irradiation had a fissile content of up to 5% U-235 or the equivalent for plutonium fuels.

The process systems used in the BNFP for the recovery of special nuclear material are an adaptation of the Purex process, for which the technology and risks are well defined. A nuclear fuel processing plant does not have high temperature systems, nor high pressure systems, nor the latent energy associated with the fissile array in a reactor core, and, except for a nuclear criticality accident, it does not have to cope with the highly radioactive short-life fission products associated with fuels being irradiated. A brief description of the process steps used in the BNFP is attached as Appendix G.

The high-level radioactive waste materials separated from the recovered special nuclear material will be stored at the site pending future disposal in a Federal repository. These high-activity wastes will be stored as acidic solutions within stainless steel tanks that are in underground vaults. Some low-level radioactivity in the form of noble gases, and vapors containing tritium, will be released from the process to the environment under controlled conditions, via a 100-

meter stack. The BNFP waste treatment system is designed to avoid the release of radioactive liquid effluents. Only nonradioactive cooling water from its heat exchangers is discharged from the plant to a canal, which carries the discharged water to Lower Three Runs Creek. The hulls from chopped fuel elements, miscellaneous fuel element hardware, and low-level contaminated solid wastes from the plant will be packaged and stored in a burial ground which will be adjacent to the facility.

The process buildings will be constructed of heavily reinforced concrete for enclosure of the critical process cells. These buildings will be located in the southwestern section of the 1730 acre site, within a plant area of approximately 14 acres. The plant area will be surrounded by a chain-link fence topped with barbed wire. Access to the plant will be controlled. In addition, the 1730 acre site boundary will be fenced.

Critical facilities are designed to withstand the forces of accidents, earthquakes or direct hit by tornado; with uninterrupted functioning of all confinement protection systems.

3.0 IMPORTANT SAFETY CONSIDERATIONS

In our review of the Barnwell Nuclear Fuel Plant, we have given special consideration to a number of site and design features which are important to safe operation and to confinement of radioactive materials. These considerations are briefly discussed in the following sections.

3.1 Site and Environment

3.1.1 Site Description and Population

The BNFP site is in Barnwell County, South Carolina, about seven miles west of the town of Barnwell. The site is owned by Allied-Gulf Nuclear Services, which will control the activities within the site. The boundary of the site is an irregular shaped rectangle approximately 1.85 miles by 1.65 miles, which contains about 1730 acres. The main process building complex and the high-level radioactive waste storage vault will be located in the southwest section of the site. The 1730-acre site will be an exclusion area, which is contiguous with the exclusion area of the Savannah River Plant (SRP) site that borders it on the north, west and south boundaries. The eastern boundary of the BNFP site, which adjoins privately owned lands, is approximately 1.25 miles (2000 meters) from the main process building complex.

The area surrounding the eastern border of the site is predominantly rural. Of the total 1,037,000 acres within a 25-mile radius from the site, there are approximately 202,000 acres of land used to produce edible crops and 15,500 acres of dairy farm

land. A significant portion of the land area is used for growing timber. There are no dairy farms within five miles of the site, but some small farms have a few cows. Approximately 800 people live within a five-mile radial zone from the plant site. Within a 25-mile radius from the site, the total population is about 57,000. Approximately 47 percent of the people live in incorporated areas. Barnwell, the nearest town, is 7.5 miles from the plant. It has a population of about 4,600. The nearest densely populated center containing more than 25,000 residents is Augusta, Georgia, which is about 31 miles from the site.

We have evaluated the radiological consequences of various postulated plant accidents with respect to potential offsite radiation doses at the site boundary and in the surrounding area. These are discussed in Section 4.0 of this report. We have concluded that an upper-limit accident in the BNFP would not exceed exposure guidelines comparable with those specified in 10 CFR Part 100.

3.1.2 Meteorology

Meteorology data were obtained from the ESSA (Weather Bureau) station at Augusta, Georgia, 35 miles northwest of the site, and from a 1200 foot television tower about 20 miles west of the SRP site. These data are applicable to the BNFP site, since there are no hills, valleys or large bodies of water in the vicinity which might invalidate the data.

On an annual basis, the predominant wind flow is from the

southwest, with wind speeds averaging about 5 meters per second, at a frequency of about 9 percent. The diffusion regimes are divided 45 percent unstable, 10 percent neutral and 45 percent stable. Precipitation in the Barnwell area is about 39 inches annually and is evenly distributed throughout the year. Heavy rainfall of short periods may be associated with the remnants of tropical storms which move inland from the Atlantic coast or from the Gulf of Mexico. Measurable snow, ice storms, damaging winds and very low temperatures rarely occur.

The BNFP is far enough inland (85 miles) to avoid most hurricane force winds associated with tropical storms along the coast. However, the most severe winds (100 mph) experienced in Barnwell County in 80 years were a result of an extreme hurricane along the coast. Nine tornadoes also were reported in Barnwell County during the period 1916-1966.

We and the Air Resources Environmental Laboratory, Environmental Services Administration, have reviewed the meteorological factors for the site, and we both agree with Allied-Gulf's analysis. The report of the Environmental Science Services Administration is attached as Appendix C. We have concluded that acceptable provisions have been made in the design of the facility to account for severe weather conditions. Structures for non-critical support services are designed to withstand the forces from 100 mph winds. Structures for essential services and process facilities are designed to withstand 300 mph wind forces from a direct hit by a

southwest, with wind speeds averaging about 5 meters per second, at a frequency of about 9 percent. The diffusion regimes are divided 45 percent unstable, 10 percent neutral and 45 percent stable. Precipitation in the Barnwell area is about 39 inches annually and is evenly distributed throughout the year. Heavy rainfall of short periods may be associated with the remnants of tropical storms which move inland from the Atlantic coast or from the Gulf of Mexico. Measurable snow, ice storms, damaging winds and very low temperatures rarely occur.

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tornado, as further discussed in Section 3.3.

3.1.3 Geology

Subsurface conditions within the BNFP area are similar, in general, to the overall SRP site conditions. Preliminary soil tests indicated, however, that an approximately 30 foot thick deposit of loose to medium dense clayey and silty fine sand, which extends to a depth of about 70 feet beneath the existing surface, was potentially susceptible to liquefaction. To increase the minimum factor of safety from 1.2 to 1.5 against possible initial liquefaction in this critical soil stratum (during a design basis earthquake condition) the shear strength in this critical soil stratum will be increased by the weight of a 15-foot high layer of compacted earth (berm) which will extend 35 feet outside the building perimeter. To consolidate the supporting soil strata, the construction schedule will permit the 15-foot earth berm to remain in place for about six months prior to starting excavations. Most settlement, if any, in the underlying soils should occur during this period. Based upon an evaluation of all samples taken from the suspect liquefaction zone, we agree with Allied-Gulf that the 15-foot earth berm will provide a conservative safety factor against liquefaction in the event of a design basis earthquake.

Several small marshy areas (locally referred to as Carolina Bays) are found within the exclusion area, the nearest being approximately 1,200 feet east of the proposed BNFP. The origin of these surface depressions is not known, but they might have been

the result of leaching of calcareous materials from the McBean Formation during a previous geologic age. This leaching may have caused subsidence of the overlying soils, which formed the surface depressions. To determine whether any calcareous sediments were present, and to ensure against the possible future subsidence of the overlying soils at the plant site, test holes were drilled to depths of 165 feet under critical sections of the proposed BNFP. No evidence of calcareous material or solution cavities was found at the building site during the drill tests.

The geology at the site was reviewed also by the U.S. Geological Survey and by John A. Blume & Associates. Their evaluations are attached as Appendices B and D.

It is our conclusion that there are no geological problems that preclude the proposed construction of the BNFP at this site.

3.1.4 Hydrology

Since the elevation of the BNFP will be at approximately 250 feet compared to an elevation of 210 feet for the roadway across the top of the Par Pond dam, flooding of the site will not occur. The topography surrounding the plant site is such that the surface water will drain toward Lower Three Runs Creek. Surface water also percolates slowly into the ground due to the flatness of the site and shallow depressions in some areas. The ground water at the plant site moves slowly to the south and southwest toward Lower Three Runs Creek. The ground water contours indicate that there is no ground water flow from the plant site to Par Pond.

Under the plant site, the mean free ground water level is about 40 feet below grade and it varies about +10 feet depending upon the season of the year. A confined zone of water which lies below the free ground water is under moderate artesian pressure.

All municipal water supplies within a 25-mile radius of the plant site utilize wells for water sources. The nearest well used for potable water is at the Barnwell Gatehouse to the Savannah River Plant. It is a shallow well about one mile east of the BNFP. The BNFP will draw water from wells in the Tuscaloosa Formation, at a depth of about 800 to 900 feet below grade, for use as cooling water. The uncontaminated cooling water will be discharged to Lower Three Runs Creek.

The flow of water in the Lower Three Runs Creek, as measured about five miles downstream of Par Pond at Patterson's Mill Bridge, varied from 5 cubic feet per second (cfs) to 500 cfs from July 1959 to August 1968. The average flow for this period was approximately 170 cfs. During the 5 cfs flow period, the flow of water over the Par Pond spillway was nil and recent rainfall had been negligible. Beyond Patterson's Mill Bridge, the Lower Three Runs Creek flows in a southerly direction for an additional 14 miles, picking up additional flow from many tributaries before discharging into the Savannah River about 140 miles above the outlet into the Atlantic Ocean.

The U.S. Geological Survey, Office of Radiohydrology, has advised us that it is difficult to predict with any certainty

the direction and rate of ground water migration at this site. Furthermore, the future pumping of ground water from the Tuscaloosa aquifer at this site may affect existing ground water hydraulic gradients and thus alter the direction and rate of ground water migration. They have recommended a more extensive mapping of the ground water table and additional studies to determine the direction and rate of ground water migration in the vicinity of the BNFP. The U.S. Geological Survey (USGS) report is attached as Appendix B.

In conjunction with the overall site environmental monitoring program, Allied-Gulf is committed to make additional hydrology investigations prior to the submittal of an application for an operating license. The additional hydrology investigations will include those recommended by the USGS, and continued observation of the ground water behavior throughout the operation of the BNFP to more accurately estimate the rate and direction of migration in the event radionuclides should be released to the ground.

Based upon our review of the proposed confinement structures and systems, we have concluded that the designs will provide acceptable protection against possible leakage of radionuclides into the soil. Furthermore, we requested and Allied-Gulf has agreed to package the solid radioactive waste containing long-life transuranic nuclides in such a manner that these containers can be recovered from the burial ground essentially free of contamination

anytime within a period of 20 years. Considering the above and the information in the PSAR pertaining to the preliminary hydrology studies and the soil characteristics at this site, we have concluded that in the event some radioactive material did escape to the ground, the ground water travel time and the soil's ion-exchange capacity will be adequate to mitigate any potential hazard to the public.

3.1.5 Seismology

There are no identified active faults or other recent geologic structures that could be expected to localize earthquakes in the immediate vicinity of the site. Based on historical seismic events in the southeastern United States, Allied-Gulf has chosen for the design bases for structures a value of VII (modified Mercalli Scale) as the operating basis earthquake, which yields a surface acceleration of 0.12g on Hershberger's (1956) curve, and a value of 0.20g surface acceleration for the maximum design basis earthquake. Our consultants, U.S. Coast and Geodetic Survey and John A. Blume & Associates, agree that the design criteria and design basis earthquake surface acceleration selected for seismic design of the BNFP are acceptable. The U.S. Coast and Geodetic Survey report is attached as Appendix C.

3.1.6 Environmental Monitoring

The DuPont Environmental Monitoring Group, Radiological Sciences Division of the Savannah River Laboratory, has maintained a continuous monitoring program of a 1200 square-mile area outside

the AEC's Savannah River Plant since 1951, which includes the BNFP site and Lower Three Runs Creek. To obtain an independent evaluation of the effect of the BNFP on the area, an environmental monitoring program will be initiated by Allied-Gulf prior to the operation of the plant. Environmental monitoring will be performed both by the BNFP staff and by qualified contractors. The monitoring program will encompass an 1800 square mile area and will include measured concentrations of radionuclides in air, water (surface and subsurface), milk, vegetation, soil and stream sediments, and radioecology studies of representative plants and animals. This program will be coordinated with the AEC's Savannah River site environmental monitoring program, the States of Georgia and South Carolina, and appropriate local health agencies.

The proposed environmental monitoring program has incorporated in it the recommendations by the USGS and the Fish and Wildlife Service and is acceptable. A copy of the report from the Fish and Wildlife Service is attached as Appendix E.

3.2 Plant Design and Radiological Safety

Bechtel Corporation will perform the detailed architectural and engineering design of the BNFP, and the Daniel Construction Company, Inc., will be the constructor.

The facility is designed to protect plant personnel and the public from inhaling, ingesting, or becoming contaminated by radioactive materials or exposed to radiation. The processing operations will be performed within the shielded cells (restricted access) in the Process

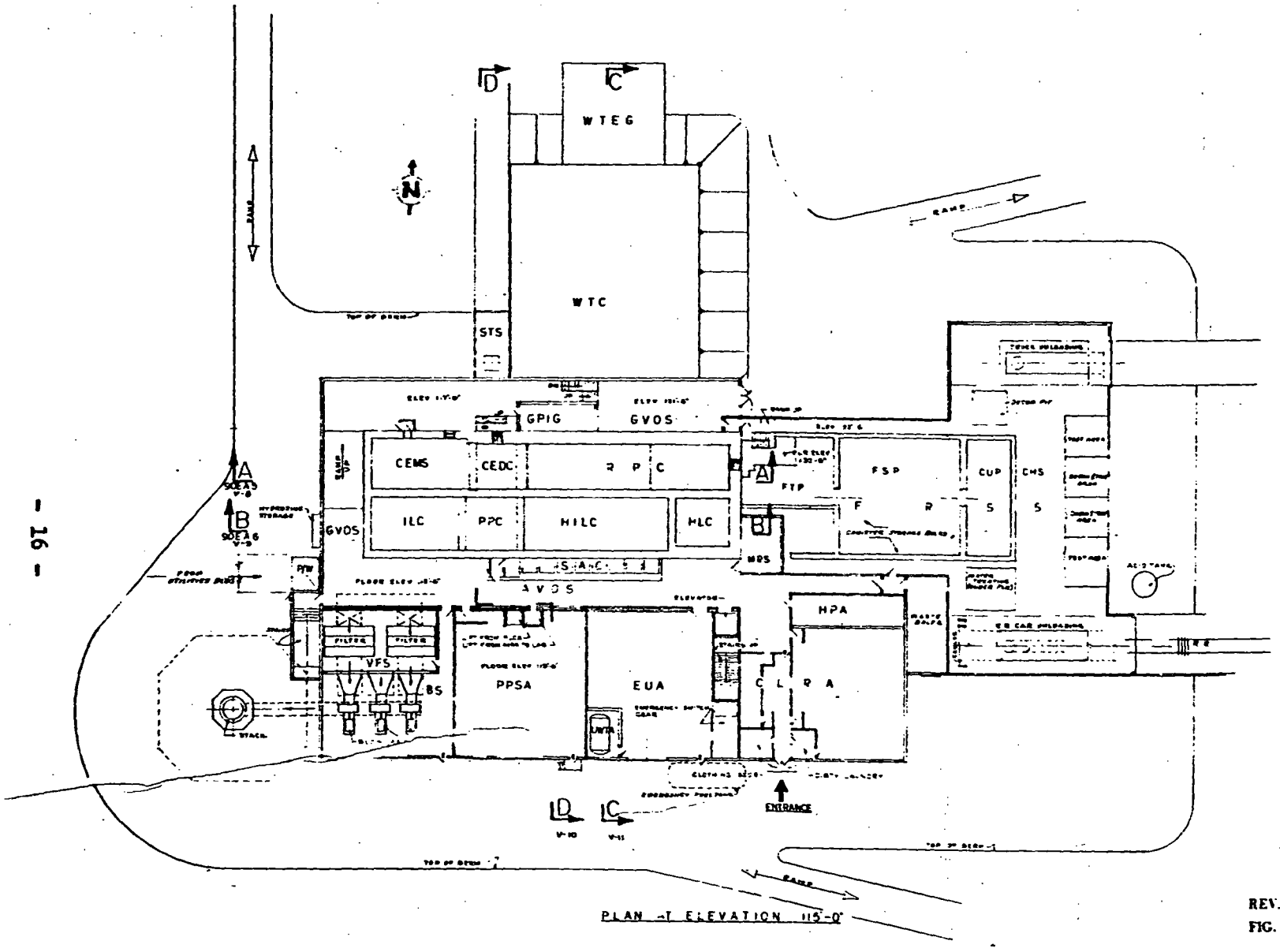

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FIG. V-2 OVERALL BUILDING PLAN ELEVATION 115'-0"

Figure 1

Building, which are shown in Figure 1. The processing operations will be controlled from outside these shielded cells by remote operation from supporting galleries (limited access), stations, areas, and aisles (normal access). A control room and emergency utilities also are provided to enable the operating personnel to perform an orderly shut down of the plant and to maintain the process inventories in a safe condition, even under accident conditions.

The structures and equipment serving as confinement barriers for radioactive materials will be designed to withstand forces resulting from accidents, earthquakes or tornadoes.

A relatively isolated site has been selected and access to the facility will be controlled. Access to the plant area will be through gates, which are locked or guarded at all times. After entering the plant area, access to the process facility will be through a change room only.

Personnel working in the plant will be provided with protective clothing and equipment. All personnel entering the plant area will be required to wear personnel dosimeters, and records of exposures will be routinely maintained. Instrumentation will be provided for detection and assessment of exposure of personnel to alpha, beta, gamma and neutron radiation, as well as detection of contamination of personnel by radioactive materials. The instrumentation to be provided include portable survey and monitoring equipment, hand-and-foot-counters, film badges, and audible and direct reading dosimeters.

Radiation detection monitors and constant air monitors will be

installed in normal and limited access areas to permit assessment of local conditions. Air sampling stations will provide a means of measuring accumulated air contamination and for determining the radionuclide(s) present. Radiation monitors will alarm locally and in the control room to indicate either high radiation levels or monitoring failures.

Respiratory protection equipment will be provided to mitigate inhalation of radioactive materials and toxic chemicals in the event of an emergency. This includes the use of respirators (primarily for non-radioactive dust or low-concentrations of non-alpha radioactive particulates), and the use of fresh-air masks and/or fresh air suits connected to breathing air stations located strategically in normal access and limited access areas, where the need can be reasonably anticipated. For emergency situations where a breathing air station is not available, self-contained, fresh-air breathing apparatus will be provided.

It is expected that during the life of the plant, some equipment failures will occur. To preclude a radiological incident, monitors are provided in the system to detect equipment failures and to automatically provide corrective action, or signal a need for prompt action by an operator. In the event of an emergency, the mobility of the radioactive process solutions permits rapid termination of process operations. If an equipment failure should occur, solutions can be transferred to other vessels. The process cells and waste storage vaults will have stainless steel pan-liners that will adequately

confine process solutions which might leak or spill from equipment in the cell. The process cells, the area under the fuel pools, and the waste storage vaults have instruments and alarms to detect leakage into their sumps. The equipment in the process cells can be repaired or replaced by remote or contact maintenance. Where it may not be feasible to repair or replace installed equipment, such as a waste storage tank, spare equipment is installed, in place, for standby use.

Process cells will have floors 3 feet thick and walls up to 5-1/2 feet thick, constructed of reinforced concrete designed for ductile behavior. These concrete structures are expected to undergo some elastic cracking if subjected to the maximum stress levels induced under earthquake loadings. However, such cracks would close up after removal of the load, and will not have any effect on the concrete's integrity as a ventilation confinement barrier, as a solution barrier, or for radiation shielding.

Most of the process vessels within cells will be designed to have the capability to withstand a design basis earthquake with respect to support of the vessels and confinement of solutions within the vessels. Failure of certain components on or in a vessel such as the internal plates, chemical service lines, or evaporator tube bundles, would not constitute a significant safety hazard provided the vessel retains its contents. Confinement of solutions within vessels will protect against possible nuclear criticality excursions or fires which might occur if the various solutions in a cell were all combined in the pan-liner. As further margin for safety, the pan-liners will be

designed to contain the inventory of the largest vessel in the cell and to maintain segregation of spillage from vessels where it might be possible for a mixing of organic solvent with a boiling evaporator solution to result in a fire or where a possible nuclear criticality accident might otherwise happen.

3.3 Design of Class 1 Structures and Components

Class 1 structures, systems and equipment are those whose failure could cause uncontrolled release of radioactivity, or those whose function is required to effect and maintain a safe plant shutdown. Class 1 structures and systems are designed to withstand combinations of loadings under normal and accident or natural phenomena conditions without loss of function. When a system as a whole is referred to as Class 1, portions not necessary to maintain essential confinement may be designated as Class 2 or 3 as appropriate. Class 2 structures and systems are those whose failure would not result in an uncontrolled release of radioactivity and whose function is not required to effect and maintain a safe plant shutdown. Roofing and siding on Class 2 structures are designed to withstand an operating basis earthquake and 100 mph wind loads. Class 3 structures and systems are those which are not essential for safe shutdown or maintenance of the plant.

The design objectives are to assure that (a) the radioactive solutions shall be retained within vessels, (b) the structures shall retain their integrity and confine radioactivity within the process cells, and (c) the emergency utility services, critical components, vital controls, and personnel shall be protected to assure that there

is no loss of a safety function pertaining to the confinement of, and control over, radioactive materials. The safety criteria provides for two or more confinement barriers to be breached in order for radioactive solutions to escape to the environment.

The 15-foot earth berm will be constructed of clayey sand which will be compacted to 95 percent of its maximum dry density in layers 6 to 8 inches by sheepsfoot or pneumatic rollers. The maximum differential settlements predicted to occur under time-lag influences of the berm surcharge loading is about 1-inch for the Process Building and 5/8-inch for the Fuel Storage Pool. All Class 1 structures will be designed for 2 times the maximum predicted differential settlement within and between structures. Foundation base mats of the critical structures will be designed as rigid elements to ensure uniform settlements.

Critical structures will be constructed of reinforced concrete, in accordance with Part IV-B, Ultimate Strength Design of American Concrete Institute (ACI) Code 318-63, wherever applicable. The Fuel Receiving and Storage Station, Process Cells and Galleries, Waste Tank Cells, Waste Tank Exchanger Gallery, and Operating and Emergency Utility Area are designed as separate structures with appropriate allowances for deflection and deformation under design basis earthquake loading conditions. The Fuel Receiving and Storage Station will be structurally separated from, but connected to, the process cell complex by a watertight joint.

The class of construction of buildings, systems and equipment are

identified in Table V-1, Section V, Volume I of the PSAR. Critical components such as product storage tanks, high-activity waste storage tanks and heat exchangers, off-gas systems, and vital process air and cooling water systems will be designed as Class 1.

Where designated Class 1 (in Table V-1), vessels and piping will conform to the applicable requirements of USAS[1/] B31.7-1968 "Nuclear Power Piping Code" for design, materials, welding, nondestructive testing and acceptance. Although paragraph 700.1.5(d) of the Code states that: "This Code does not apply to piping for nuclear installations designed or used specifically for processing nuclear fuels," Class 1 vessels and piping in the BNFP will conform to the applicable portion of this Code in all parts except those portions involving high temperature - high pressure confinement (Section 1) to assure conservative designs.

The plutonium storage tanks will be fabricated with corrosion resistant stainless steel having wall thickness minimum of 1/4" and constructed in accordance with the applicable portions of USAS B31.7-1968 code as described above. The high-level radioactive waste storage system tankage and piping also will conform to the applicable portions of USAS B31.7-1968 as described above.

Vessels not designated Class 1 will meet the American Society of Mechanical Engineers (ASME) Section VIII, Unfired Pressure Code for pressure vessels. Those vessels not operated as pressure vessels will

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1/ United States of America Standards Institute Code.

meet the reference code requirements, but will not require the ASME Code Stamp. Fabrication and quality verification will be consistent with ASME Section VIII, Unfired Pressure Code, USAS B31.1.0-1967, or pertinent ASTM standards and specification.

The design basis for the piping external to the process cells that conduct high-level radioactive liquid waste to and between the high-activity waste tanks will be double-walled pipe constructed of corrosion resistant stainless steel and designed to conform to USAS B31.7-1968, where applicable.

All piping entering or leaving structures will be designed with the necessary flexibility to withstand the maximum differentials in seismic response between soil and structures, and between structures, without rupture or loss of function.

All piping, vessels, and equipment serving as confinement barriers for radioactive materials will be constructed of corrosion resistant stainless steel or materials that are resistant to or compatible with the contained solutions.

The seismic design criteria for the BNFP are based upon a repitition of the 1886 Charleston Earthquake, which is the most severe earthquake recorded in this area. In the vicinity of the BNFP, the intensity of such an operating basis earthquake (OBE) would be about VII, with a corresponding ground acceleration of about 12% of gravity. Under these conditions, the BNFP is intended to be capable of remaining in operation while proceeding with an orderly and safe shut down. For the design basis earthquake (DBE) the design basis is an intensity of

VIII at the site, with a corresponding ground acceleration of about 20% of gravity. Under such conditions, the BNFP is intended to retain its integrity to assure a safe shut down and controlled confinement of the radioactive materials.

We agree with the OBE and DBE intensities selected for the seismic design bases. However, we believe that the response spectra to be used for the seismic design of the process building, the high-level waste storage facilities, and the critical equipment and piping, should reflect the greater amplification in the four records obtained for the earthquake model suggested by Allied-Gulf to represent what might be expected at the BNFP in the event of an earthquake. Therefore, we proposed that Allied-Gulf modify the initially proposed design bases response spectra for periods above 0.2 seconds by averaging the four records obtained from the suggested earthquake model and then normalizing to the design bases for seismic ground motion under OBE and DBE conditions. We believe also that the time-history generated acceleration spectra should not fall below the above modified design bases response spectra in the critical range between 1 and 20 cps. Allied-Gulf has agreed to make adjustments in accordance with our recommended modifications to the design bases response spectra. For design purposes, vertical ground acceleration will be considered to be two-thirds of the horizontal ground motion. The damping factors to be used in the dynamic analyses will be:

	Percent of Critical Damping	
	OBE	DBE
Concrete Shear Wall Buildings, as a whole, with energy lost due to soil-structure interaction	5.0	9.0
Reinforced Concrete Shear Walls and Elements	2.0	4.0
Welded Steel Frames	1.0	2.0
Bolted Steel Frames	2.0	4.0
Piping with Rigid Supports	0.5	1.0

A time-history analysis will be performed to generate response spectra for use in the design of critical equipment, appendages and piping. Such acceleration spectrum, scaled by computer, will approach, but will not fall below the design response spectrum in the critical range between 1 and 20 cps.

We and our consultant agree with the design basis response spectra and the proposed damping factors to be used. The evaluation of seismic design criteria by John A. Blume & Associates, our consultant, is attached to this report as Appendix D.

The BNFP is designed also to withstand a direct hit by a tornado, with all safety-related systems and components continuing to function. The BNFP design criteria for tornado wind loading on all Class 1 structures are:

1. Uniform loading for any particular wall element will be based upon a 300 mph wind velocity, and

2. Uniform loading due to depressurization (3 psi in 3 seconds) of vented structures will be added algebraically to the portion of the total uniform wind loading due to the 300 mph wind velocity applied as positive on the windward side and negative on the leaward side, in accordance with American Society of Civil Engineers (ASCE) Paper #3269.

The design basis for exterior walls will be seventy-five percent of the ultimate strength capacity of the concrete for flexure, bond and shear, based upon ACI 318-63, formula 16-1.

The venting of Class 1 structures containing radioactivity will be effected through the ventilation system, which will be designed to withstand tornado effects. Those areas in Class 1 structures that do not contain radioactivity will be vented by means of blow-out panels or louvers.

Class 1 structures (except roofs) will be designed to resist the following loadings:

1. A 4" x 12" x 12' long wood plank striking on end at 300 mph from grade to elevation of the structure.
2. The maximum wind velocity (300 mph) combined with a 4" x 12" missile striking end on at 200 mph, and the maximum pressure differential (3 psi in 3 seconds).
3. The maximum wind velocity (300 mph) combined with a 4000-pound automobile striking the building at 50 mph (flying through the air at elevations up to 25 feet above ground) with a contact area of 30 square feet, and the maximum pressure differential.

The roofs on Class 1 structures will be designed for the combined effect of the 4" x 12" missile striking on end at 200 mph, the maximum wind pressure, and the maximum pressure differential. The 4" x 12" missile velocity is based upon velocities which might be attained (a) by an object being thrown from the core of the tornado, in relation to the low height of the BNFP structures, or (b) by a fall from a higher elevation. The service areas above the process cells and the control room provide additional protection against falling debris.

Critical safety related equipment, instrumentation, electrical service, and piping, which may exist in or traverse Class 2 or 3 zones, will be fully protected from tornado wind and missile damage by Class 1 structural protection provided locally in the form of walls and roof elements. Underground piping will be covered with a depth of earth adequate to provide protection against tornado-generated missiles.

We have concluded that the design criteria and design bases for tornado loading and missiles, which are defined above, are acceptable.

Based upon our review of the process systems, we have concluded that the structures, components and piping that are designed and constructed to the above criteria will ensure adequate confinement of radioactive solutions.

3.4 Confinement Systems

3.4.1 Fuel Receiving and Storage Station

The Fuel Receiving and Storage Station (FRSS) will receive and unload irradiated fuel from shipping casks transported by either truck or rail, store the irradiated fuels in multielement

storage canisters prior to processing, and prepare the emptied casks for return shipment. When the cask is received, it is transferred by a 150-ton over-head crane into the Cask Unloading Pool (CUP), which is about 14' x 28' x 55' deep. Fuel elements are removed one at a time from the cask and are inserted into a multielement storage canister, which can hold 4 PWR or 6 BWR elements, by a 5-ton cask unloading crane. The fuel storage canister is then transferred by a canister crane to the Fuel Storage Pool (FSP), where it is secured in a storage rack. Neither the shipping cask nor the 150-ton over-head crane traverse the Fuel Storage Pool. The Fuel Storage Pool, which can store about 180 MTU, is about 48' x 48' x 28' deep. From the Fuel Storage Pool, the fuel storage canister is moved to the Fuel Transfer Pool (FTP), where one element at a time is removed from the fuel storage canister, by a canister unloading crane, and transferred to the shear in the Remote Process Cell.

The fuel elements themselves provide confinement of radioactivity. Damaged fuel elements will be canned, if necessary, when they are removed from the casks before being placed in the fuel element storage canister. Storage canisters containing damaged fuel elements will be stored in a separate area, which has an underwater hood that will collect gaseous releases and vent them to the vessel off-gas system. The canisters and racks are designed (a) to withstand a design basis earthquake, (b) to protect against the possibility of accidental criticality in the Fuel

Storage Pool, and (c) to provide protection against an accidental drop of a fuel element storage canister.

The pool water cooling system initially will remove up to 5 million Btu per hour of decay heat. The design includes provisions for future installation of additional heat exchanger equipment to handle twice this heat load. The fuel pool water will be maintained between 70° and 120° F, and between a pH of 6.5 to 8.5. A water treatment system will be provided to maintain water clarity, and to limit radioactivity levels in the pool water to less than 2000 dpm/ml beta-gamma and 1.0 dpm/ml alpha, on the average. At these levels, <1 mr/hr, there will be no required restrictions on the working time in the pool areas.

The fuel pools are Class 1 structures. The cranes and the FRSS building frame and crane rail supports also are Class 1. The walls and roof enclosing the FRSS, however, are Class 2 structures. The latter is acceptable since all piping, equipment and other conceivable missiles that could originate in this building, or in adjacent areas, will be anchored to withstand a tornado wind loading. The fuel elements will be handled and stored under at least 12 feet of water. The water provides protection against impact by missiles. The racks and canisters will be designed also to resist the impact loadings due to potential missiles. A direct hit by a tornado would not result in loss of water sufficient to cause a loss of cooling capability or a significant reduction in radiation shielding.

The Cask Unloading Pool (CUP) and the Fuel Transfer Pool (FTP) will be lined with stainless steel. The interior of the Fuel Storage Pool (FSP) will be covered by an acid and radiation resistant paint. Each pool will be isolated by gates that can be inserted between the pools for additional protection against the loss of water in the event of an accident in the Cask Unloading Pool. An external drainage and a confinement barrier made of Volclay panels, gravel and clay are provided to back-up the base slabs and walls of the fuel pools. The external confinement barrier is designed to preclude the loss of water from the fuel pools, which might occur in the event of damage to the concrete barriers. If there is any leakage from the fuel pools it will drain to sumps, and will be returned to the pool by automatic pumps that are capable of returning leakage back to the pools at a rate of 100 gallons per minute (gpm). In the event of an emergency, water can be added to the fuel pools at a rate of 500 gpm.

Based on the design information provided, we have concluded that the CUP can cope with a cask drop incident, since the liner and confinement barriers would restrict both the rate and the amount that the pool water level could drop, which might otherwise result in high radiation exposure to personnel in the area. The liner and back-up confinement systems also would limit the leakage rate sufficiently to permit gates to be installed (for an added margin of safety) between the CUP and FSP, and, if necessary, makeup water can be added at a rate of 500 gpm to maintain adequate

shielding. We also have concluded that the leakage collection and confinement systems surrounding the fuel pools are acceptable for protecting against seepage of contaminated water to the soil.

Based upon our evaluation of the information submitted by Allied-Gulf in the PSAR, we have concluded that the design criteria and the design bases are acceptable for the Fuel Receiving and Storage Station.

3.4.2 Process Building

The Process Building has approximate outside dimensions of 88 feet wide by 170 feet long by 82 feet in height above the bottom of the foundation. The foundation will consist of a single, continuous, rigid mat covering the area of the entire structure. The structure will be founded on natural soils at a depth of 14 feet below natural grade, i.e., about 29 feet below the finished grade of the berm fill.

The Process Building houses a complex of process cells containing the equipment for processing the radioactive materials. These process cells will be surrounded by stations and support areas arranged to provide confinement capability consistent with operating and accessibility requirements. In general, reinforced concrete construction will be used for process cells and to house vital protection systems such as the emergency utilities and the central control room (Class 1 structures). Other process support areas (Class 3 structures) will be constructed with structured steel frames, insulated metal siding or concrete walls, and

concrete or metal deck with insulation and built-up roofing.

Within the Process Building there will be remote and contact maintenance process cells, which are located in two rows, side by side, sharing a common shielding wall. These cells will be enclosed by thick (3 feet to 5-1/2 feet) reinforced concrete, and will have stainless-steel pan-liners. The process cells will encompass an area of about 50 feet by 150 feet and will be about 63 feet deep. The following is the nomenclature and general arrangement for the main process cells:

(see Figure 1)

Remote Maintenance Cells	Contact Maintenance Cells
RPC-Remote Process	HLC-High Level
CEDC-Crane & Equipment Decontamination	HILC-High Intermediate Level
CEMS-Crane & Equipment Maintenance (Station)	PPC - Plutonium Process
	ILC - Intermediate Level

The Remote Process Cell (RPC) contains the fuel element shear, the dissolver, the HA centrifugal contactor, and the high-activity waste concentrator. The operations in the RPC can be observed through two shielded viewing windows in both the Grade Viewing Operating Station (adjacent to the RPC) and the Top Viewing Operating Station (above the RPC). The RPC is contiguous with the CEDC, which is separated from the CEMS by a heavy shielded door that can be raised. Remote operations and maintenance work will be performed in the latter two cells with two remote bridge cranes and a power manipulator. Radioactive-solid wastes are

packaged in, and removed from the plant, via the CEDC.

The High Level Cell (HLC) will contain the equipment for process feed to the HA centrifugal contactor, and the equipment for the denitration of the high-activity waste prior to sending it to the waste storage tanks.

The High Intermediate Level Cell (HILC) will contain the off-gas chemical treatment equipment, the solvent extraction equipment for partitioning plutonium from uranium, the equipment for treatment and recycle of the first organic solvent system, the General Purpose Evaporator, and the equipment for concentration of low-activity waste.

The Plutonium Process Cell (PPC) will contain the equipment for the first and second cycle plutonium purification systems, and vessels for plutonium storage and load-out.

The Intermediate Level Cell (ILC) will contain the equipment for the second cycle uranium extraction system, for silica gel treatment, for recovery of nitric acid, for treatment and recycle of the second organic solvent system, and the waste solvent burner.

Allied-Gulf evaluated a process explosion with respect to possible effects on the confinement capability of filters, hatches, viewing windows, the pool waterseal and ventilation ducts. The analyses determined that an explosion typical of the kind that could occur in the process cells would not affect the confinement capability of the BNFP.

We have concluded that the building process cell arrangement, and the grouping of components, provides acceptable isolation and confinement to mitigate potential accidents. We agree that the likelihood of a process explosion, and its magnitude, would be low and would not significantly affect the confinement capability of the process cell complex. We agree with design criteria requiring process vessels and critical piping within the process cells to survive a DBE, and with the pan-liners as back-up for collection of maximum leaks from a single equipment failure that might result from corrosion or an accident.

3.4.3 High-Activity Liquid Waste Storage

The high-activity liquid waste, which contains the bulk of the fission products from the processing of irradiated nuclear fuels, will be a self-heating liquid. Allied-Gulf proposes to store the high-activity wastes as a nonboiling nitric acid solution (1.0 M to 5.0 M) to allow for its future conversion to a calcined solid form for transfer to a Federal repository. The storage of high-activity liquid wastes as a nonboiling nitric acid solution is based upon established engineering principles which have been demonstrated by about 15 years of experience to be acceptable.

Allied-Gulf proposes to provide initially three high-activity liquid waste storage tanks, including one spare. The initial waste storage system will have a total capability to store up to 270,000 gallons of nonboiling high-activity waste solution. Each waste

tank will hold up to 135,000 gallons of waste solution and will contain cooling-coils to remove fission product decay heat at rates of up to approximately 29 x 10^6 Btu-hr^{-1} from each waste storage tank. The three high-activity waste storage tanks will be contained in a stainless steel lined concrete vault which is buried under about 10 feet of earth. A separate waste tank and a spare tank will be provided, in a separate vault, for the interim storage of the radioactive liquid waste bottoms from the General Purpose Evaporator. Process piping to and from all waste storage tanks will be contained within secondary barriers. The waste tanks and vessels associated with the waste cooling system will be vented to the vessel off-gas (VOG) system. The related structures will be vented to the Process Building ventilation system via the Remote Process Cell. All structures and components for the storage of radioactive liquid wastes will be designed as Class 1. Protection system instrumentation will be designed to conform to the Institute of Electrical and Electronics Engineers, Inc. (IEEE) Criteria No. 279.

The high activity liquid wastes will be maintained at a nonboiling temperature of about 140°F. The maximum temperature will not exceed 160°F at the maximum design basis heat load of 29 x 10^6 Btu-hr^{-1} for each waste tank. The fission product decay heat will be removed from each waste tank via a closed-loop, primary cooling system, which includes six sets of cooling-coils within the waste storage tank, a surge tank, a circulation pump,

and a heat exchanger. The decay heat is transferred in the heat exchanger to cooling water supplied from the multiple deep wells. Uncontaminated cooling water is discharged to the canal leading to the 15-million-gallon water-cooling pond and then to Lower Three Runs Creek.

The design of the waste tank cooling system includes, in addition to the three tanks and related cooling systems, two spare heat exchangers, two spare surge tanks, and two spare circulation pumps. These additional components are provided to assure a capability to supply adequate cooling to the high-activity liquid waste under normal and accident conditions.

As a further backup to the normal and emergency spare cooling system described above, Allied-Gulf has added two diesel-driven pumps, each of which is capable of supplying an adequate supply of cooling water directly to the primary cooling loops in the high-activity waste storage system by recirculating water from the 15-million-gallon water-cooling pond. The two diesel-driven pumps, the piping and the structure will be designed Class 1. This system will be designed so that either pump can be started remotely from the control room. However, the diesel pumps for recycling cooling-pond water will be used only if all of the other emergency systems do not operate, and for only that interim period required to restore the other cooling system to operation.

The piping in the high-activity liquid waste storage system will be fabricated from 304L stainless steel according to the

appropriate sections of USAS Code B31.7. The recirculation pumps, including the spares, will be connected to the emergency power system. The primary cooling water loop will be monitored for radioactivity, and the liquid levels in the waste tank and the level in the primary loop surge tank will be monitored also for detection of a cooling-coil failure. An air sparge system will be used to keep solids from settling in the waste tanks. Sparger air also is used to maintain the radiolytic hydrogen concentration in the tank vapor space, which is monitored, below two volume percent. The vault sumps will be instrumented to detect the presence of liquid, and the air in the vault will be sampled to detect radioactivity, which would indicate a leak in the waste storage or cooling system. Two independent methods for transferring the sump's contents to the waste tanks, and for emptying waste tanks, such as air lifts and steam jets, will be provided.

The waste storage tanks and the waste storage vault liners, piping and heat exchangers will be fabricated according to specifications and quality controls to ensure a high integrity confinement system. Corrosion tests with simulated waste solutions, similar to the composition of the waste solutions anticipated from the BNFP process, are being made by Allied-Gulf. Corrosion tests will continue during plant operation to ensure the continued integrity of the waste storage system and the vault liners.

A highly reliable water supply and cooling system is required to assure confinement of the high-activity waste solutions under all conditions. (The cooling water supply system is discussed in Section 3.7.4.) If cooling water is interrupted for several hours, these waste solutions could self-concentrate, which could result in a release of volatile and semi-volatile fission products to the vessel off-gas system and eventually to the atmosphere. Further, it could result in an accumulation of precipitated radioactive solids which could cause hot-spots on the tank's surface that could result in a breach of confinement. However, before the waste solutions reach this condition, it will be possible to establish emergency cooling water supplies, to dilute the waste solution to preclude the precipitation of solids, to switch to alternate spare cooling systems, or to transfer the waste to the spare tanks. All of these alternatives will be immediately available to cope with loss of cooling water.

The closed loop cooling system (primary and secondary heat exchange) provides assurance that the cooling water effluent which is discharged to Lower Three Runs Creek will be uncontaminated water. However, in the event the primary cooling water is contaminated and if a tube in the secondary heat exchanger should fail, the leak will be detected by gamma monitors on the cooling water effluent line and by analysis of composite samples of the cooling water effluent. The leak detection instruments will sound an alarm in the control room and the cooling water effluent from

the heat exchanger will be automatically diverted to the retention reservoir. (This protection system is discussed in Section 3.7.6.)

Pending development and implementation of AEC policies pertaining to the ultimate disposal of high-activity wastes, Allied-Gulf has defined only the initial storage of the high-activity waste solutions. The initial capability proposed for storing high-activity liquid wastes will be adequate for confinement of the wastes generated during the first and second years of plant operation. It will be necessary for Allied-Gulf to provide additional storage capability for the high-activity liquid waste, or to convert the high-activity liquid waste to solids for transfer to a Federal repository, prior to filling the second waste storage tank. The safety considerations related to any future design modifications or any future changes with respect to the method for storage of high-activity waste solutions will be subject to review and approval by the staff.

Based upon the design criteria and the design bases in the PSAR, we have concluded that the design of the proposed interim high-level radioactive waste storage system for nonboiling acid waste solutions will provide acceptable redundant features to assure confinement of the high-activity liquid wastes. We have concluded that Allied-Gulf's proposed system for the interim storage of high-activity waste solution is acceptable.

3.4.4 Ventilation Systems

The Fuel Receiving and Storage Station, the Process Building, and the control room each has independent ventilation systems. In addition, the process vessels are vented to an off-gas treatment system which provides for removal of radioactive iodine and the removal of nitrogen oxides. These systems have high-efficiency filters which remove radioactive particulates from the ventilation exhaust streams. All ventilation systems will be in continuous operation and will exhaust to the atmosphere.

The Fuel Receiving and Storage Station receives inlet air (27,000 cfm) at the top level of the building, which is washed and conditioned, and distributed to the cask unloading air lock, to the testing and service area, to both cask decontamination areas, to the fuel storage pool area. The ventilation air from this building is released directly to the atmosphere. Radioactive gases which might be released from damaged fuel elements will be vented to the vessel off-gas (VOG) system via underwater hoods which collect the gaseous releases from storage canisters containing damaged fuel elements.

The Process Building ventilation air is supplied by one blower (86,500 cfm) located at the top level of this building. The inlet air is washed and conditioned. Fresh air is supplied directly to the control room. Offices, and other normally occupied areas will be air conditioned.

The Process Building supply air is distributed via ductwork equipped with air diffuser and volume-balancing dampers. The

supply air is first introduced into the normal access zones at a positive pressure and then flows to the adjacent limited or restricted access zones, in the direction of increasing radioactivity, which are at lower pressures. Frequently used access openings between limited access zones and normal access zones will be provided with air locks. Pressure adjustments, between the outside and the various zones, will be manually balanced to give a 0.1 to 2-inch range of water differential to assure that air flows follow prescribed paths within the building.

The pressure differential across each zone barrier will be measured, and an alarm will sound in the control room if there is as much as 0.05 inch of water change in the pressures between zones. The response to such alarms will be a manual adjustment of the air flow.

Air from the office areas, the change room, the lunchroom, and the laboratory area will be discharged directly to the atmosphere. Air from the laboratory's hoods that handle low-level radioactive samples will be filtered and discharged from a small stack on the roof of the analytical laboratory. Air from the shielded analytical cells will be filtered and discharged from a small stack on the roof of the analytical laboratory. Air from the shielded analytical cells will be filtered and exhausted to the main building ventilation system. Air that was not discharged directly to the atmosphere from normal access areas will be exhausted, after filtration, by three 50-percent capacity fans (37,000 cfm) to the stack. These

exhaust fans are connected to the emergency supply system. To prevent over-pressurizing the building, the air inlet supply blower is interlocked with the exhaust units, which must be in operation before the supply blower can be energized, and pressure sensors will sequentially operate to automatically start the spare exhaust fan and stop the supply fan.

Backflow dampers will be installed only on the discharge side of the three exhaust fans.

We have evaluated the building ventilation system, and have concluded that it is acceptable. The proposed three-zone control system is simple and reliable. It does not depend upon complex damper systems, nor backflow dampers, which might be difficult to adjust and maintain. The limited access zone is an acceptable physical barrier, which separates the normal access areas from the restricted access areas. If there should be an explosion in a process cell, the limited access zone will mitigate ventilation flow reversal and will confine any related spread of contamination to the limited access and restricted zone.

The dissolver off-gas (DOG) and the vessel off-gas(VOG) treatment systems provide for extensive removal of radioactive contaminants (except noble gases and tritium) and removal of the nitrogen oxides prior to the discharge of gases from the process system to the stack. The off-gas vapor from the acid fractionator will contain the bulk of the tritium released by the dissolution of the fuel elements.

The DOG system will receive from the shear and the dissolver gaseous discharges that contain primarily krypton, particulates, iodine, and nitrogen oxides. The DOG system will consist of a condenser, a vapor-liquid phase separator, a mercurous-mercuric nitrate-nitric acid scrubber for the removal of greater than 99 percent of the iodine and particulates, and an absorption column where nitrogen oxides will be oxidized with air and absorbed in water. At this point the DOG stream will be combined with the VOG stream for additional cleanup. The DOG system will be designed to maintain a 4 -10" H_2O vacuum in the vapor space of the dissolver during dissolution.

The VOG system will receive gases from shipping casks, from any leaking fuel elements, from the high-activity waste tanks and from the various process vessels. After the VOG stream has passed through a condenser it will join with the DOG stream. The combined DOG and VOG streams will pass through a mercurous-mercuric nitrate-nitric acid scrubber followed by a silver impregnated inorganic adsorption bed. Then the combined stream will flow through a pair of high efficiency filters, installed in series, before being discharged by a steam jet to the stack. A blower (2000 cfm) with knock-out pot will be provided to operate as an emergency backup exhaust system. The VOG system will be designed to maintain a 2-7" H_2O vacuum in the vapor spaces of the process vessels.

The DOG-VOG stream will be monitored, prior to entering the ventilation exhaust stream, by instrumentation similar to that on the stack.

The overheads from the acid fractionator will be condensed, collected and analyzed for tritium and gross beta-gamma activity to determine whether the condensate is within acceptable limits for revaporization and release as a vapor to the stack. The condensate from the General Purpose Evaporator also will be revaporized and released as a vapor to the stack. Control limits, which will govern the amounts of radioactivity that may be in the condensate to be revaporized and released as a vapor to the stack, will be incorporated in the technical specifications of the operating license.

Gas fired heaters are used to provide a capability during an emergency to maintain the off-gas system at adequate temperatures until the process system is shutdown, and then off-gas heating would not be required.

The process-gas treatment system is based on demonstrated technology, and we have concluded that the proposed off-gas treatment system is acceptable.

The ventilation ducts between the cells and the cell filters, and between the cell filters and the main ventilation filters will be Class 1. The ventilation duct from the Process Building to the stack, and the stack itself will be Class 1 structures. All ducts in restricted access areas and in the main exhaust system will be stainless steel. The cell ventilation systems will be designed to have adequate durability to remain operable at all times.

Radioactive particulates will be removed from the ventilation exhaust streams by high-efficiency filters (HEPA) located in filter

niches beyond the discharge vents from each process cell. Multiple filter units in parallel and/or in series will be provided as necessary. The cell discharge air will be collected in a header and flow to final high efficiency filter units (two 70 percent capacity filter units), which will be located adjacent to the stack. The final filters and blowers are protected by a Class 1 structure.

The filters (HEPA) will contain one roughing stage or include one high-efficiency stage in series, and will be replaced semi-remotely or by contact maintenance. The fiberglass filters, which are fire resistant, are designed to operate at temperatures up to 350°F. They will remove 99.97 percent of 0.3 micron diameter particles. The HEPA type filters are designed to withstand a pressure loading of 10 inches of water gage across the entire face for short periods without permanent damage to the filters. These filters are capable of surviving shock pulses at about 3 psi for 50 milliseconds.

The ventilation systems have filter protection features including:

a) Differential pressure indicators and alarms on the off-gas treatment system to alarm if there is a pressure buildup due to accumulation of particulate on filters or liquid in the iodine adsorber.

b) Monitors to measure the buildup of gross beta-gamma activity on the DOG-VOG filters.

c) Pressure sensors on the dissolver vent header, vessel vent header and at the VOG filter outlet, which will

cause the Emergency VOG blower to go automatically into operation if any of these pressures begin to build up higher than normal, and

d) Differential pressure indicators and alarms on the cell filters, laboratory filters and final filters to monitor their condition at all times.

The performance of filters is also evaluated continuously by the radiation monitors in the gaseous effluent streams.

The filters are protected from possible damage by fires and explosions by the duct system's pathway and an adequate distance between the process cells and the final filters to protect against the effects from heat and blast.

The ventilation system, including the filters, will be under continuous surveillance and can be maintained to acceptable standards. We have concluded that the proposed designs are acceptable for confinement of radioactive particulates under normal and accident conditions. As an added margin of safety, during an emergency, the release of radioactivity could be confined within the building by shutdown of the ventilation system and by closing the dampers beyond the exhaust fans. This action would, of course, contaminate parts of the Process Building; however, it would confine radioactive particulates largely within the restricted access and limited access zones of the building.

3.5 Radioactive Effluents and Solid Waste

The BNFP will release gaseous effluents to the enviornment via a

stack which exhausts to the atmosphere about 100 meters above natural grade. Solid wastes containing known or detectable amounts of plutonium and transuranic isotopes will be stored in a burial ground located adjacent to the BNFP. The BNFP will not discharge radioactive liquid effluents to Lower Three Runs Creek.

3.5.1 Gaseous Effluent

Radioactive gaseous effluents from the BNFP process will be chemically treated and/or filtered prior to being discharged to the atmosphere about 100 meters above grade, and will be continuously monitored to record the quantity of radioactivity discharged from the stack. The radioactive gaseous effluents will consist primarily of (a) krypton 85 from the dissolver off-gas, (b) tritium which will be released as revaporized condensate, (c) iodine 129 and 131 which will be chemically treated to remove greater than 99 percent from the off-gas stream, and (d) residual amounts of minute particles having alpha, beta, and gamma activity, which might pass through the filtering system.

A constant air monitoring system, with recorded and on line readout and alarms, will be provided for the detection of gross beta activity. The lower-limit detection capability will be about 1×10^{-10} uCi/cc for cesium 137 and 9×10^{-6} uCi/cc for krypton 85 in the stack effluent. A representative sample of the stack effluent will be collected continuously. The composite sample will be analyzed, at intervals no greater than 24 hours during operating periods, for iodine 131, ruthenium 106 and major fission products

and transuranic particulates.

In addition to monitoring the overall stack gas, the dissolver (DOG) vessel (VOG) and acid fractionator (AOG) off-gas will be monitored separately. The release of krypton 85 will be determined by collecting grab-samples from the DOG and performing laboratory analyses for krypton 85. Tritium releases will be determined by collecting representative samples from the acid fractionator condensed overheads and performing laboratory analyses for tritium.

The isopleths of concentration, based upon the meteorology data integrated over one year, indicate that the maximum annual relative concentration (X/Q) at ground level will occur within the exclusion area between 1000 and 2000 meters from the BNFP stack. The maximum annual relative concentration at the Barnwell site boundary (2000 meters) would be approximately 9×10^{-8} sec. m^{-3}. Our estimates of maximum annual off-site releases of radioactive gaseous effluents are compared to the upper-limit air concentration values in 10 CFR Part 20 in the following table:

MAXIMUM AVERAGE ANNUAL OFFSITE CONCENTRATION
(Normal BNFP Operation)

Radioisotope	BNFP Stack Release Rate (Ci/sec)	Offsite Max. Conc. (Ci/m^3)	Appendix B Table II, Column 1 10 CFR Part 20 (%)
Kr-85	0.47	4.3×10^{-8}	14.3
H-3	1.9×10^{-2}	1.7×10^{-9}	0.9
I-131	6.7×10^{-7}	6.0×10^{-14}	<0.1
Particulate			
Fission Products	2×10^{-6}	1.8×10^{-13}	<0.1
Transuranium	1×10^{-7}	9.0×10^{-15}	0.3

The quantity of radioactivity released by the Savannah River Plant to environs in the Barnwell area is, for the most part, too small to be distinguished from natural background radiation or is obscured by fallout.[1/] Under normal conditions, detectable SRP radioactive effluents in the atmosphere at the Barnwell site would be below one-tenth percent of the values in 10 CFR Part 20, Appendix B, Table II, Column 1. Therefore, because the radioactive effluents from the SRP are relatively insignificant amounts, and since these plant operations are not related, we have evaluated the Barnwell Nuclear Fuel Plant as an independent facility.

Based on our review of the treatment and filtration system and the effluent control and radioactivity detection capability of the BNFP, we have concluded that releases will be within 10 CFR Part 20 limits and that, within the current state of technology, Allied-Gulf has designed reasonable systems for limiting radioactive effluent quantities and concentrations to as low as practicable in the environment. The design of the BNFP will include provisions for future installation of equipment to remove noble gases from the effluent. The noble gases (principally krypton 85) will be extracted from plant effluents when practical systems have been demonstrated for subsequent long term storage of this gas in a Federal repository. Practical systems for the removal of tritium from the vapor effluent currently are not available.

1/ See Semiannual Reports, Effect of the Savannah River Plant on Environmental Radioactivity.

3.5.2 Solid Waste Storage

Contaminated equipment, fuel element hulls, and miscellaneous fuel element hardware and radioactive-solid waste will be suitably packaged, or decontaminated, prior to being transported to a low-level waste burial ground for storage in an area adjacent to the BNFP. The solid waste will be buried, at depths above the water table, in a manner that will minimize the percolation of rain water through the soil in the vicinity of the buried waste. As long as the soil remains unsaturated, there would be negligible, if any, leaching of radioactivity in the ground water.

Baskets of hulls and other dissolver scrap will be routinely monitored until conditions are established in fuel chopping and dissolver operations that ensure complete dissolution of special nuclear material. After such operating conditions are established, a statistical analysis will be used to spot check whether the fuel chopping and dissolver operations continue to be satisfactory with respect to adequate removal of special nuclear material from the hulls. After complete dissolution of special nuclear material, the leached hulls will be dumped remotely into cylindrical reinforced concrete containers, which will be capable of holding the hulls from about six metric tons of uranium. Filters and other solid wastes that might contain plutonium will be packaged with or in a manner similar to that for hulls. The storage canisters will be sealed and then will be transferred on a shielded trailer to the burial ground. At the burial ground, the storage container will

be removed from the shielded trailer via a shielded or remotely operated crane and placed upright in the burial ditch, which will be backfilled by a shielded earth mover.

The dissolution, acid-leach and water wash time cycles in the dissolver will remove practically all of the special nuclear material from the hulls. After exposure to the dissolution, acid-leach and water wash cycles, it is not likely that transuranic residue, if any, would be soluble in water. Further, the proposed packaging of solid waste in concrete storage containers will mitigate leaching by water and the proposed construction of the burial ditches will mitigate the migration of radionuclides in the ground. The proposed packaging method also will allow for removal of such buried solid-waste if in the future it might be desirable to do so.

We have concluded that the proposed method for storage of the radioactive-solid wastes is acceptable. We will require a technical specification governing the dissolver operations and the monitoring of hulls to assure that the quantity of residual special nuclear material in the hulls will be as low as practicable.

The low-level radioactive solid burial ground at a nearby site, which will be operated by another company, is not a part of this licensing action.

3.6 Process Instrumentation and Control

The chemical process used in the BNFP for the recovery of special nuclear material is an adaptation of the Purex process, which has been

used in AEC and other fuel reprocessing plants. The experience gained over the prior 15 years has identified the safety considerations associated with this process, and these are adequately defined in the PSAR. Allied-Gulf proposed to use acceptable methods, which are based on established practices in similar facilities, for protection against accidents related to the handling of chemicals and metals in the BNFP.

New technology is being developed by Allied-Gulf to accomplish the partitioning and stripping of plutonium in the co-decontamination and plutonium purification cycles. The new technology and related equipment has been demonstrated only in the laboratory and further development work is in progress to demonstrate its reliability in a nearly full-scale pilot plant test. However, the radiological safety considerations associated with Allied-Gulf's choice of these unit operations are the same as those related to alternative unit operations that are currently used in the Purex process to accomplish the partitioning of plutonium from uranium, and the stripping of plutonium product streams.

The BNFP process systems are discussed in Section III of the PSAR, and a brief summary of the process is attached as Appendix G. On the next page is a simplified block flow diagram of the BNFP process system.

The BNFP process systems are closely coupled. To keep the inprocess inventory of radioactive solutions as low as practicable, the BNFP has few intra-cycle process vessels. Process control is maintained by the use of the usual instruments and controls that are used in related automated industrial chemical plants. However, the

BNFP

PROCESS SYSTEM

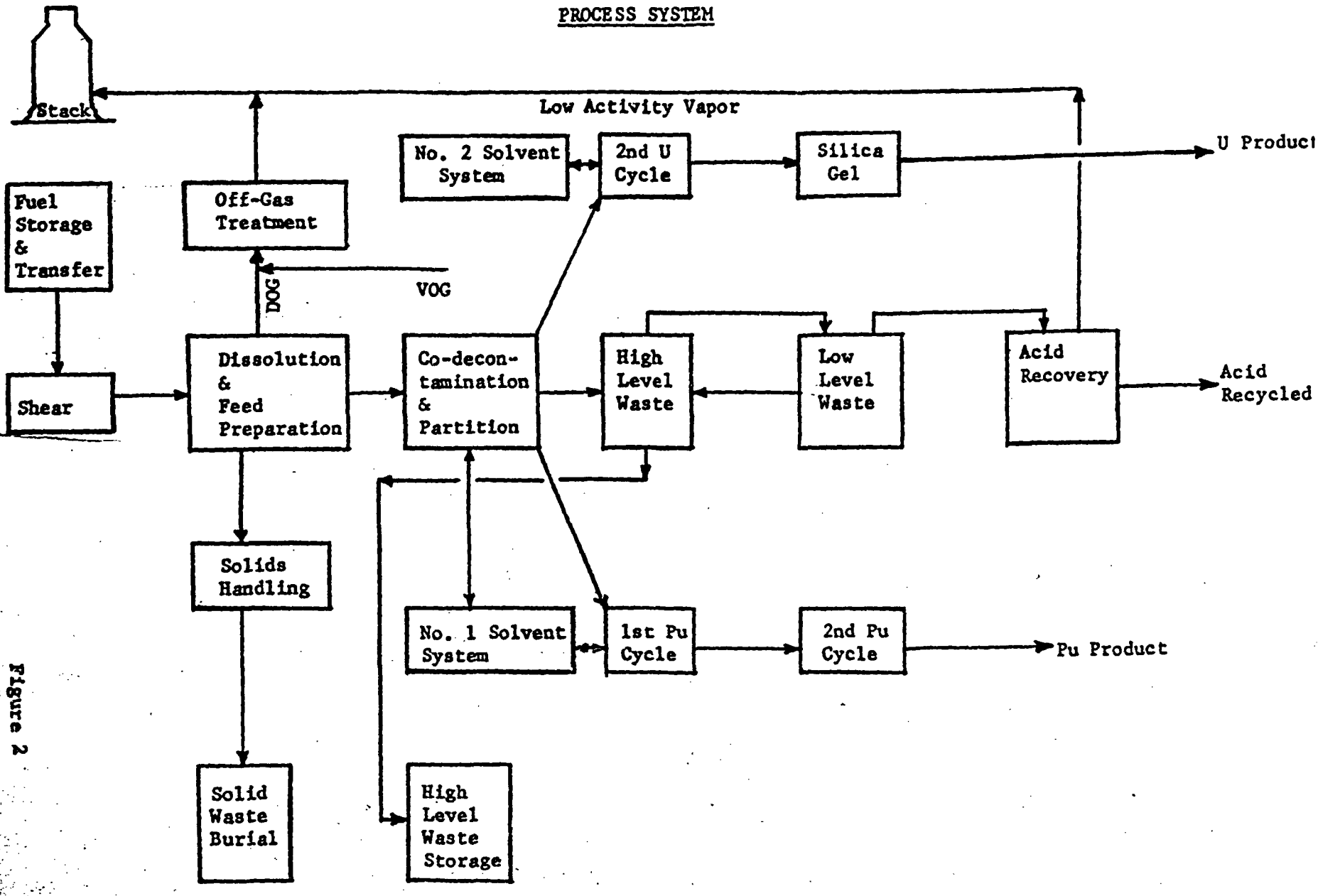


Figure 2

processing of irradiated nuclear fuels requires additional special precautions and safety features to provide protection against possible process-related nuclear accidents. The nuclear criticality considerations and the additional instruments that are provided for an added margin of safety to protect against possible-process related accidents are discussed below.

3.6.1 Nuclear Criticality Protection

In Table II in Section 7, Addendum No. 2, to the PSAR, Allied-Gulf presents in tabular form a preliminary criticality evaluation of the proposed design bases that will assure that the facility can be operated safely with respect to protection against nuclear criticality excursions during both normal and abnormal process conditions. Criteria also are presented by Allied-Gulf for determining safe interaction between arrays of fissile containing units. Solution transfer control, leakage from process vessels, and control of precipitants in systems whose nuclear safety is based on homogeneous solutions, have been considered in other sections of the PSAR.

Whenever practical, the determination of safe design and operating parameters will be based on experimental results. Whenever experimental results are not directly applicable, design and operating parameters will be defined by performing a detailed reactivity analysis. The bases for such analyses will be a comparison to systems of known reactivity, which are similar to those under consideration regarding enrichment, composition, moderation,

size, configuration and use of neutron absorbers. Administrative and process controls are based on the double contingency principle, i.e., no single credible equipment failure or human error can cause a criticality incident.

We have reviewed the nuclear safety design bases and criteria for the BNFP and we have concluded that they are acceptable for prevention of criticality incidents under foreseeable combinations of circumstances that might occur in the BNFP. In general, the criteria covers geometry control, concentration (moderation) control, and the use of both soluble and fixed neutron absorbers, which are based on accepted principles for nuclear criticality protection. Detailed designs and criticality analyses of all equipment, assemblies, accessories, etc., where criticality safety is assured by geometry or fixed neutron absorbers, will be submitted for our review prior to the actual fabrication of these components.

3.6.2 Dissolution

The safety related considerations in this step of the process are (a) protection against nuclear criticality excursions, and (b) the control of the chemical dissolution rate, which in turn governs the rate of release of effluents via the off-gas ventilation system.

Before introducing fuel elements into the process, each fuel element will be identified to ensure that the proper elements were selected for processing with respect to cooling time, fissile contents and economic considerations. The chopped pieces of the fuel

element (1-inch to 5-inch lengths), which fall by gravity through a chute from the shear to the dissolver, will be retained within the dissolver in a 30-inch i.d. basket. The dissolver will be filled with nitric acid containing a neutron absorber (cadmium) before the shear is started. An interlock system is provided which is designed to preclude operation of the shear unless the dissolver basket is in position and the dissolver is filled to the proper level with nitric acid containing the neutron absorber. Dissolution of the special nuclear material is carried out at about 90°C. The temperature of the dissolver solution is controlled by cooling water (containing a nuclear absorber) which is circulated in the jacket surrounding the dissolver. Chopped fuel and nitric acid are added to, and solution is withdrawn from, the dissolver on a semi-continuous basis to control the rate of chemical reaction for each batch. It will take about 3 to 4 hours to dissolve one fuel batch containing about 1,000 kg of uranium in the basket.

The content of nuclear absorber in the acid stream entering the dissolver is monitored by two independent instruments, which control an interlock system between the shear and the nitric acid stream entering the dissolver. If either monitor fails to detect an adequate concentration of nuclear absorber, the interlock system will terminate the operation of the shear and shut off the nitric stream and thus automatically terminate the dissolution operation. To prevent polymerization of the plutonium, the acid concentration of the dissolver solution is maintained above 1 molar

and is continuously monitored. The pressure in the vapor space above the solution, the solution density and the liquid level are monitored also. If any of these variables exceed control limits, the monitoring instrument automatically terminates the operation of the shear, which in turn will terminate dissolution.

Based upon our review of the design criteria and the design bases information provided in the PSAR, we have concluded that the proposed shear and dissolver are acceptable. The designs are based upon accepted principles for protection against nuclear criticality accidents. We have also concluded that the two independent BF_3 counter and neutron source instruments (both of which must detect the presence of the proper concentration of cadmium nitrate in the liquid streams entering the dissolver in order for the shear to operate) will provide adequate assurance that the nuclear absorber is present before and during the addition of chopped fuel to the dissolver. The instruments and alarm systems for control of the process parameters are based on accepted practices that are adequate for protection against process-related accidents.

3.6.3 Co-decontamination and Uranium Purification

In the co-decontamination and partition cycle, the bulk of the fission products and the heavy elements will be separated from the uranium and the plutonium, and then the plutonium will be separated (partitioned) from the uranium, after which each product is purified separately. The purification system for uranium consists of two solvent extraction cycles (including the co-decontamination cycle) and a silica gel treatment. The bulk of the fission

products and the heavy elements other than uranium and plutonium will be in the waste (HAW) stream leaving the HA centrifugal contactor.

Process-related safety consideration are the prevention of possible explosions in the concentrators and fires in the solvent extraction system. If "red oil" accumulated in a concentrator, it would be possible under certain conditions for an explosion to occur. However, to detonate the "red oil" the solution would have to be over-concentrated to such an extent that its boiling point would exceed 135°C. (See Section 4.4.3) To prevent the over-concentration of solutions, the concentrator will be equipped with dual specific gravity and temperature control instruments, with alarms. To prevent possible fires in the solvent extraction system, the operating temperatures in equipment containing the organic solvent will be maintained below the flash point of the solvent.

The fissile material in the feed stream (HAF) will be made safe by concentration control and association with the nuclear absorber added in the dissolver. Prior to processing in the solvent extraction systems, the composition of the HAF stream will be positively identified by sampling and analyses. In the subsequent uranium purification system, nuclear criticality protection is based largely upon the control of the concentration of the fissile material by the composition and the relative flow ratios of the process streams. Instruments, with alarms, warn the process

operator if variables deviate from acceptable conditions. For an additional margin of safety, neutron monitors are provided on the HS column and the IC column which will provide warning of the accumulation of fissile material that might occur due to internal plutonium recycle or failure of plutonium to partition in the IB(C). A monitor on the HAW stream will provide warning also of excessive amounts of plutonium and uranium which may appear in the waste stream. The flow controllers installed on the scrub and strip streams have low-flow alarms to alert the operator if flow rate adjustments are necessary to prevent unsafe accumulation of fissile material. In addition, the low-flow alarms on (a) the plutonium organic recycle stream, (b) the HS column's scrub stream, and (c) the IB(S) column's strip stream, and the neutron monitor on the IB(C) column, will automatically terminate the feed stream (HAF) if corrective action has not been taken by the process operator within 10 minutes. If the monitor on the IBU stream detects plutonium being carried over into the uranium purification cycle, it will automatically recycle that stream back to the partition cycle and terminate the streams flowing to the HA/HS contactors. The uranium product stream has a gamma-activity monitor to provide warning if this stream is above limits for safe handling in unshielded areas.

Based upon our review of the information provided in the PSAR, we have concluded that the process controls and the protection instrumentation identified above provide control over process

variables with adequate margins of safety for protection against process-related accidents. The proposed protection instruments are acceptable because (a) routine process analyses will verify whether the detection instruments for fissile material are functioning properly, (b) the alarms on flow controllers and the automatic response features will be routinely tested, and (c) the interaction of process control instruments will provide adequate indications of process conditions in the event of a single instrument failure. The technical specifications in the operating license will specify how many instruments must be functioning properly in order to continue the operation of a process system while instrument repairs are being made.

3.6.4 Partition and Plutonium Purification

In the partition cycle, the plutonium valence state is reduced from +4 to +3. The plutonium in the reduced valence state will strip to the aqueous phase leaving the uranium in the organic phase. The plutonium is then oxidized back to the +4 valence state for subsequent solvent extraction. After the co-decontamination cycle and partitioning from uranium, the plutonium will be purified in two similar solvent extraction cycles.

Since protection against a nuclear criticality incident in the plutonium purification systems is based upon the safe geometric design of the system's components, the principal safety considerations in this section of the process are related to the process

streams that are recycled back to other sections of the process.

The operating ranges for process variables will be within those limits of acidity, temperature, and concentration for which plutonium solutions are known to be stable. To assure adequate control of the process: (a) the acid concentration in the scrub stream will be monitored and an alarm will sound if low acid con-concentrations are detected, and (b) the flow controllers on process streams will have low-flow alarms that alert the operator if there is a need to make corrective adjustments to maintain the established flow rate. The organic streams leaving the plutonium purification cycle (2BW and 3BW) are monitored for plutonium content and (a) if high in plutonium, these streams are recycled back to their respective "A" columns, or (b) if low in plutonium, these streams are recycled to the HA centrifugal contactor. The monitor on the 2BW or 3BW will automatically select the recycle routing to protect against an excessive amount of plutonium being returned to the co-decontamination and partition cycle. The final plutonium product stream will be monitored for gamma-activity to warn the operator if this stream is above limits for safe transfer from shielded to unshielded areas.

Each aqueous waste stream from the plutonium purification cycles is monitored for plutonium content and (a) if the plutonium content is low, the waste stream will be routed to the LAW concentrator, or (b) if the plutonium content is high, the waste stream

will be recycled. The LAW concentrator bottoms are returned to the HA centrifugal contactor for recovery of residual plutonium from the waste streams and to combine these wastes with the high-activity waste stream.

The plutonium purification system has a separate condenser and knock-out pot on its vessel vent system to mitigate the spread of plutonium to the VOG system.

Based upon our review of the information provided in the PSAR, we have concluded that the above protection features are acceptable for providing an adequate additional margin of safety to the normal process control and instrumentation systems.

3.6.5 Neptunium Recovery and Purification

During the campaigns to recover and purify neptunium, the plutonium and uranium bearing streams will be processed in the same manner, under the same controls, and in the same equipment as under normal conditons. Neptunium is not a fissile material.

3.6.6 Product Handling

Uranium - Uranyl nitrate product solution will be transferred from a product storage tank to a weigh tank, for accountability, prior to transfer to a tank truck. The decontaminated uranium solution will have a known concentration of fissile material and its concentration will be below the limits required for criticality in an infinite medium.

Plutonium - The plutonium nitrate product will be transferred from geometrically safe product storage tanks into geometrically

safe product storage tanks into geometrically safe plastic bottles. This operation will be performed within a shielded, ventilation enclosure. The decontaminated plastic bottles in turn will be weighed for accountability, and then will be placed in a shipping container via an exit port in the enclosure. The shipping containers, which will conform to AEC and DOT regulations, will be transferred to rail car or truck for shipment from the site.

In the event plutonium product storage is provided as a customer service, such storage will be in geometrically favorable tanks that may contain fixed neutron absorbers. In this case, prior to shipment the plutonium product will be treated, if necessary, to remove americium. This will be done by passing acidified plutonium product solution through a cation exchange column, which will remove americium and trace quantities of iron. Following concentration, the plutonium product will be packaged as described above. The americium will be recovered and transferred to shielded containers, or sent to the high-activity waste storage tank.

Neptunium - The load-out of neptunium will be handled in the same enclosure, and in the same manner as the plutonium product.

3.6.7 Solvent Treatment

To maintain the TBP-hydrocarbon solvent purity necessary to achieve satisfactory results with respect to final product purity specifications, radioactive and other impurities are continuously removed from the solvent by washing it alternately in alkaline and

nitric acid solutions and by filtration.

Continuous uranium monitors on the first alkaline washer in each system will automatically divert the carbonate wash solution to a holding tank if a buildup of high uranium concentrations is detected in this equipment. The diverted wash solutions will be acidified and recycled to the feed adjustment tank.

Organic liquid wastes will be burned in a device similar to a domestic oil-burner. A flame detector will automatically turn off the organic feed pump if the solvent fails to ignite or the flame goes out. Safety devices also will prevent introducing solvent to the combustion chamber while the chamber is at elevated temperature, and shut down the unit if fumes fail to ignite within a reasonable time during startup.

We have concluded that the process instrumentation and controls for the solvent treatment system are based on acceptable practices which have been demonstrated to be adequate.

3.6.8 High-Activity Waste

The HAW stream from the HA centrifugal contactor, which contains the bulk of the fission products, will be reduced in volume, by a factor of 20-30, in the HAW concentrator. After concentration, the nitric acid content of the waste will be reduced by chemical reaction with sugar and then the waste will be transferred to the high-activity waste tank for storage.

Temperature control interlocks will be provided on the steam and the cooling water, which will be designed to prevent the HAW

concentrator solution temperature from approaching 135°C and to prevent self-concentrations due to the heat from fission product decay. The steam pressure also will be limited to a maximum of 25 psig, which corresponds to a saturated steam temperature of 130°C. In this system, cooling water also will be required for temperature control. In addition, provisions are provided for the manual addition of water, for diluting and cooling the contents of the concentrator, if necessary, to cope with abnormal process conditions.

Based upon our review, we have concluded that the instruments and alarms are adequate to maintain process control and to warn the operator of abnormal conditions.

3.6.9 Low-Level Waste and Acid Recovery

The concentration of low-level radioactive liquid wastes will be carried out in two separate systems, one for acid process wastes and one for all other low-activity wastes. Both systems will use continuous thermosyphon evaporators which have bubble cap trays and high efficiency mist eliminators for the removal of particulate matter from overhead condensates.

The LAW (Low Acid Waste) concentrator will reevaporate the overhead condensate from the HAW (High Acid Waste) concentrator and evaporate all other acidic raffinates from the process system. The LAW concentrator's overheads will be condensed and sent to the Acid Fractionator for recovery of nitric acid, and its bottoms will be sent to the high-activity waste system via the HA contactor.

The LAW normally will not contain significant quantities of fissile materials. However, the LAW concentrator feed tank will contain a fixed poison in the form of boron glass Raschig rings, which will provide adequate protection against criticality in the event that abnormal amounts of plutonium are inadvertently transferred to this tank from the 2AW and 3AW Diversion Tank. In addition, a monitor for fissile material will be installed on the LAW concentrator, which will sound an alarm if the accumulation of fissile material in the concentrator becomes significant. The introduction of organic materials into the LAW concentrator will be minimized by decanting in the evaporator feed tank. The temperature of this concentrator also is controlled below 135°C to preclude a "red oil" explosion.

Prior to being fed to the Acid Fractionator, the overhead from the LAW concentrator will be continuously monitored, and recycled if necessary, to prevent the introduction of entrained radioactive materials into the Acid Fractionator. The fractionator will produce a concentrated nitric acid product (approximately 12 M), which will be recycled to the various parts of the process system.

The bulk of fission product tritium will be contained in the overhead water vapor from the fractionator. This vapor will be condensed, sampled and analyzed for tritium and gross beta-gamma activity. The condensate normally will be revaporized and discharged to

the stack downstream from the final filters. If the beta-gamma activity of the condensate is above the acceptable limits specified in the technical specifications, the condensate will be decontaminated by using it in place of the scrub water normally supplied to the Acid Fractionator.

The General Purpose Evaporator (GPE) will be used to concentrate miscellaneous wastes, such as contaminated cask water, cooling water and alkaline wash solutions used for solvent treatment. The bottoms from the evaporator, which will contain fission product activity, will be stored in a separate tank within a high-activity waste storage vault pending conversion to an acceptable solid form for disposal in a Federal repository for high-activity waste. The GPE overhead vapor will be condensed, sampled and analyzed, and will be combined withthe condensate from the Acid Fractionator.

We have concluded that the proposed systems for handling low-level radioactive waste solutions, i.e., storing the bottoms for future disposal in a Federal repository, and releasing the decontaminated overhead vapor under controlled conditions to the atmosphere via the stack, are acceptable. The low-activity effluents are decontaminated by two stages of evaporation before being released as a vapor to the stack.

3.7 Protection Systems

Confinement of radioactive materials requires that the following systems or services be maintained under all conditions: (a) effective

ventilation barriers, (b) a continuous supply of cooling water for decay heat removal, (c) a continuous supply of air to pneumatic instruments and to move solutions if necessary and (d) a continuous supply of electrical power to instruments, compressors, pumps, and blowers, and a limited supply of power to perform operations associated with the dissolver and the fuel handling cranes. All emergency protection features depend upon a reliable source of electrical power in order to perform their safety functions. For this reason, the power supply and instrumentation required to assure the performance of safety functions are redundant. The design of these protection systems will conform with the provisions of IEEE No. 279, "IEEE Criteria for Nuclear Power Plant Protection Systems".

The BNFP supplies its own utility services, except for offsite electrical power and natural gas. In the event of an interruption in the supply of offsite electrical power, emergency diesel-driven generator units will be activated by an under-voltage relay acting through a time-delay to override momentary voltage dips and power interruptions. Diesel fuel will be stored onsite, and bottled gas will be available to meet requirements for emergency operations.

3.7.1 Electrical Power

Electrical power will be supplied to the BNFP by the South Carolina Electric and Gas Company, via a single 115 kV line, 8-1/2 miles long, from their Uerquhart-Fairfax line. The total connected supply is 5000 kW, which is almost twice the normal power demand required by the BNFP. Allied-Gulf proposes to initiate immediately

a shut down of the process if the offsite power is interrupted.

Standby emergency electrical power will be provided by redundant circuits from two 1250 kW diesel-driven generator units, each of which can supply adequate power to meet the electrical power needs under emergency conditions. One of the two generators will start automatically, if the normal electrical supply voltage drops below 70% of normal for more than two seconds. In that event, the vital loads will be time-sequentially transferred to one of two emergency buses as follows:

+3 seconds - Start one of three analytical laboratory exhaust blowers.

+21 seconds - Start one of three building exhaust blowers.

+29 seconds - Start Emergency VOG blower.

+42 seconds - Start one of two air compressors.

+47 seconds - Start one of two process vessel jacket water pumps.

+50 seconds - Start DC chargers.

+50 seconds - Start diesel fuel pump.

If the first generator fails to prove itself within 13 seconds, the second generator will automatically be energized in +20 seconds and will follow the above sequence after a 20 second delay. Either generator can be started manually from the control room if necessary.

Electrical requirements for staring the emergency generator(s), maintaining "no break power" to safety-related electronic instrumentation, and emergency lighting will be provided by batteries which

will be maintained fully charged. The diesel generators will be started by high-pressure air stored in cylinders. They also can be started manually in sufficient time to avoid a safety hazard. Automatic startup of the emergency generator will restore essential electrical service within about one minute.

Based upon the design criteria stated in the PSAR, we have concluded that the emergency power generation units and a redundant electrical distribution system designed to the provisions in IEEE 279 are acceptable. Such a system would be capable of supplying adequate emergency power to meet the needs of the following essential components which supply the services that are required to assure adequate confinement of the radioactive materials in the BNFP:

1. Communications and Instrumentation
2. Lighting
3. Process building exhaust fan
4. Analytic exhaust fan
5. Emergency VOG blower
6. Well water pumps
7. Jacket water pumps
8. Air Compressors
9. Fuel Pool recirculating pumps
10. Fire water pump
11. Diesel fuel pumps
12. Cask-handling crane
13. Dissolver

3.7.2 Building Ventilation

To assure adequate continuous ventilation of the Process Building, the electrical power supply to three exhaust blowers is provided by redundant circuits. In the event of a failure in the offsite electrical power supply system, one of these three blowers, each

capable of drawing 50% of the normal ventilation air flow, will be automatically started by the emergency power supply system. The other two blowers also are supplied with emergency power and either one also can be started manually from the control room. However, only one is programmed to be restarted during the period of an offsite electrical power outage.

We have concluded that the proposed emergency ventilation design bases are acceptable. If there should be a loss of offsite electrical power, the ventilation supply blower will shut down so that the building could not be pressurized. An adequate pressure differential can be maintained between ventilation zones if one out of the three exhaust blowers is started within several minutes after the offsite power outage. Along with the termination of the offsite power, the supply of steam to evaporators would be interrupted and the process would be shutdown. During and after the shutdown of process equipment, 50% of normal ventilation air flow will be adequate for maintaining confinement.

3.7.3 Vessel Off-Gas Ventilation

Emergency back up to the DOG-VOG steam jet is provided by a blower, which can provide adequate ventilation of the vessels. The blower is supplied with electrical power via redundant electrical circuits tied into the emergency power system. The emergency blower will start automatically upon sensing a buildup in the VOG exhaust pressure.

Gas fired heaters are used in the DOG-VOG system to preclude

the condensation of vapors during process operations. The heaters are not required to operate when the process is shutdown. To cope with an interruption of normal gas services during process operations, an adequate supply of bottled gas will be kept in reserve to accomplish the shutdown of the process.

If the emergency off-gas blower fails to start when an offsite power outage occurs, some radioactivity might vent to the process cells. This would be acceptable because even if the vessels vent to the cell, the contamination will be confined within the facility by the building ventilation filters. Therefore, we have concluded that a single blower, as back-up to the off-gas jet, is acceptable.

3.7.4 Water Supply System

A continuous supply of water must be assured for cooling the high-activity waste tanks and for fire protection. The water supply and distribution systems are Class 1.

Water will be supplied from the Tuscaloosa aquifer by wells on the site. The Tuscaloosa aquifer is an adequate source for the supply of cooling water. The water system will include the process system supply; boiler and pool supply; process, fuel pool and waste storage cooling; and fire fighting and sanitary requirements. Water for fire fighting will be stored on site. Process, fuel pool and waste storage cooling will normally be a once-through cooling system. Uncontaminated cooling water is discharged to the canal leading to the 15-million-gallon water-cooling pond and then to Lower Three Runs Creek.

Emergency power will be provided to the deep well pumps to supply water for boiler makeup, fire fighting and to maintain an adequate flow of water to the waste storage tank, fuel pool and process vessel heat exchangers, where loss of cooling could jeopardize confinement of radioactive materials. At least four wells will provide cooling water, when the BNFP is operated at its design rate.

The proposed protection system provides for automatic sequential start of deep well pumps to assure that at least one of the four wells supply water. The design also provides for separate routings of the cooling water via a loop principle which includes redundant routes to assure the water supply to the High-Level Heat Exchangers. For an added margin of safety, the water distribution system is monitored by instruments to detect low-flow rates and high-flow rates, which provide warnings of pump and pipe failures so that prompt action can be taken to cope with such contingencies.

As a further backup to the redundant cooling system described above, Allied-Gulf has added two diesel-driven pumps, each of which is capable of supplying an adequate supply of cooling water directly to the primary cooling loops in the high-activity waste storage system by recirculating water from the 15-million-gallon water-cooling pond.

We have concluded that the proposed water supply and distribution systems for assuring a continuous supply of cooling water are acceptable.

3.7.5 Air Supply System

Compressed air is used to operate the pulsers, air-lifts, jets, maintenance tools, and the process instruments. The instrument sparge and transfer - air compressor is spared and both are supplied with emergency power via redundant circuits. The compressed air system is designed with Class 1 components and piping. In addition, the utility air compressor also will be supplied with emergency power, and if necessary, this compressor can be connected to supply air for instruments, spargers and for solution transfers. A separate compressor and piping system is used to supply air for pulsers. The breathing air used in contaminated work areas is supplied by an independent pipe system that is designed to protect against the possible contamination of this air supply. The compressors can receive emergency power from either diesel generator.

The redundant air supply system is designed to assure adequate air for both normal and emergency conditions. The piping distribution system itself, which is a Class 1 system, is not redundant. However, in the event of a pipe failure in any specific section of the plant, it will be possible to repair the break or improvise an alternative means to supply air to monitor specific vessel contents or to move solutions from a cell. Furthermore, the probability of a break in an air line is low, and an air line failure to or in a cell is an acceptable risk since a single failure would not directly affect safety nor the confinement of radioactive material in a cell.

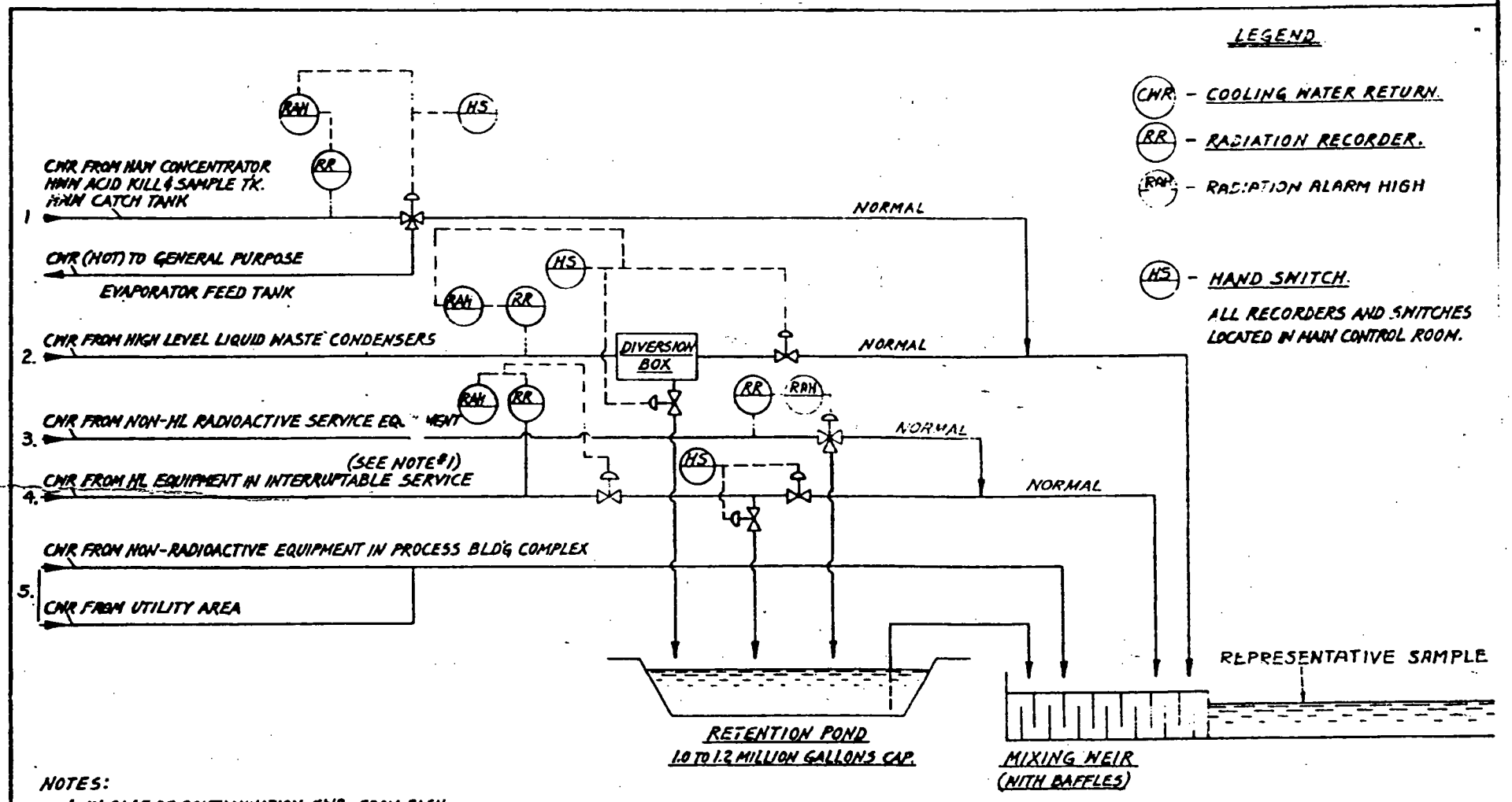
Air rather than steam will be used as a motive force where a continued supply would be essential for safe shutdown. During normal operation, steam will be used to drive turbines, for jet movement of process fluids, for the VOG exhaust jet and for building and process heating requirements. Steam will be generated in one of two normally gas fired boilers.

We have concluded that the compressed air supply and distribution system are acceptable.

3.7.6 Heat Exchanger Effluents

During normal operation, the BNFP will not release radioactive material in the liquid effluent. However, in the event of a leak in a heat exchanger, the effluent could become contaminated. A simplified schematic of the normally non-radioactive liquid effluent discharge paths is presented in Figure 3 on the next page.

The cooling water and steam are at pressures greater than the process solutions being heated or cooled so that if a leak develops the flow will be into the process system rather than into the liquid effluent discharge path. However, it may be possible under certain conditions for minor amounts of radioactivity to seep through pin hole leaks or hair line cracks that could result from corrosion. In that event, radioactivity might enter the cooling water or steam condensate. For this reason, the liquid cooling water and condensate effluent paths are monitored by redundant instruments and by routine analyses for radioactivity.



NOTES:

1.- IN CASE OF CONTAMINATION, CWR - FROM EACH INDIVIDUAL SERVICE WILL BE STOPPED UNTIL LEAK IS ISOLATED & CORRECTED

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J. MOHRBACHER
2-18-70

SCHEMATIC DIAGRAM
LIQUID EFFLUENT
RADIATION MONITORING
SYSTEM.
FIG. X-3

DWG. BY
E. BERRY
1-12-70

Figure 3

If a leak develops, a radiation detection alarm will sound, and automatic controls will (See Figure 3): (a) divert Stream 1 to the General Purpose Evaporator, or (b) divert Streams 2 and 3 to the retention basin and switch to an alternate heat exchanger, or (c) interrupt the cooling water flows to Stream 4 temporarily, until the leak can be isolated. The contaminated water diverted to the retention basin (from streams 2, 3 or 4) will be decontaminated by ion exchange prior to being released to Lower Three Runs Creek.

Commercially available detectors are not sufficiently sensitive to detect radionuclides such as Sr-90 and Pu at the low concentration limits in 10 CFR Part 20 but, since other radionuclides (Ru-106 and Zr-Nb-95) also would be present, the detection alarms on liquid effluents will sound at levels of activity nearly at MPC.

In the event of a leak in a heat exchanger, the diversion of the liquid effluent to the retention basin, which is capable of holding one-million gallons for further treatment, is adequate backup to assure against a significant accidental release of radioactivity to Lower Three Runs Creek. Furthermore, the design bases and codes provide for materials, methods of construction and quality assurance testing to assure a low probability that such leaks could occur. Therefore, based upon our review and evaluation of the proposed systems for treating and controlling the release of possible contaminated liquid effluents to the environment, we have concluded that the proposed protection system is acceptable provided

that Allied-Gulf performs routine analysis of the composite effluent sample on a schedule to be established and defined in the technical specifications.

3.7.7 Fire Protection

Fire fighting systems for normal industrial type fires will consist of automatic sprinklers, fire hose stations, and portable extinguishers. A foam system will be used in process cells for controlling fires of a chemical nature. For controlling fires of metallic nature, such as zirconium fires, Metal-X will be used in the Remote Process Cell.

The site will be provided with a water distribution system, consistent with National Bureau of Fire Underwriters standards, which will include water storage, a motor-driven pump, an engine-driven spare pump, fire hydrants and fire hose stations. Fire-proof materials will be used whenever possible to minimize fire hazards within the plant.

Temperature-sensitive devices and/or smoke detectors will be installed in process cells and ventilation ducts. These will automatically release a fire extinguishing medium and actuate visual and audible alarms at the central control panel.

Based upon the results of tests of commercially available fire detectors (Fenwal Report No. PSR-366), we have concluded that fire detectors in strategic locations in the process cells will be able to detect a fire and respond by releasing foam within about

three minutes. Based upon the results of tests of commercially available smoke detectors (Fenwal Report No. PSR-368), we have concluded that certain smoke detectors are reliable and can respond within about three minutes.

4.0 ACCIDENT ANALYSES

4.1 Criteria for Chemical Processing Plants

The siting guidelines in 10 CFR Part 100 apply to stationary power and testing reactors. Except for a nuclear criticality accident, upper-limit accidents in a nuclear fuel processing plant release transuranic and fission product particulate rather than the radioactive gases. Particulate releases would be confined largely to the site. To evaluate whether the proposed site is acceptable relevant to the risks associated with postulated accidents in the BNFP, we have used an upper-limit guideline that constitutes a degree of hazard comparable to the accident dose guidelines in 10 CFR Part 100, which is:

> An amount of radionuclides equal to the time-integrated inhalable concentrations for 50 years of exposure to the airborne concentrations shown in Table II, Column 1 of 10 CFR Part 20, i.e.:
>
> TID max.[1/] $= \overline{MPC}_a$ $(Ci/M^3) \times 1.58 \times 10^9$ (Sec.)

Based on the above criteria, the guideline value for the release of plutonium from the BNFP stack is limited to about 84 (A+B) curies. A person in the area of maximum ground level exposure (within the exclusion area, about 400 meters from the stack) during the time of such a release could receive, in a single intake of plutonium, a bone dose lifetime commitment of about 150 rems. Persons beyond the site boundary, in the Barnwell area, could receive no more than 55 rem bone dose lifetime commitment. Similar guideline values were established for the other

[1/] Maximum Total Integrated Dose

radionuclides that also could be released from the BNFP stack. However, in the event of an upper-limit accident the transuranic radionuclides become the limiting guideline values.

4.2 Accident Experience

During the 20 years that nuclear fuel processing plants have been in operation, the frequency and severity of accidents have been significantly lower than that in the related chemical industry. Only a few accidents involving radioactive material have occurred, and none resulted in damage offsite. The experience gained from these few accidents has resulted in improved engineered safety features and operating procedures in nuclear fuel processing plants, such that the probability is low that similar accidents would occur in the future. Nevertheless, for the purpose of evaluating the site and the protection inherent in the BNFP design we have postulated major accidents, which represent the upper-limit accidents that could occur in the BNFP. For each postulated accident we have assumed simultaneous failure of process safety features (an unlikely event) and have assumed the worse probable consequence. Our assumptions and our evaluations are discussed below. If such major accident could occur, there probably would not be a release offsite.

4.3 Assumptions

We made the following assumptions which govern our evaluation of the possible consequence of upper-limit accidents:

* An explosion would disperse solution containing soluble radionuclides into the cell's atmosphere. Radionuclides, up to the amount involved

in the accident or that contained in a quantity of solution equivalent to a heavy mist (100 mg. solution/m^3) in one cell volume of air (whichever is limiting), would be exhausted to the cell's ventilation system.

This assumption is conservative since air containing a heavy mist would retain only about 10% of such particles after impinging against a wall or baffle.

* All of the iodine and 0.1% of the ruthenium associated with the contents of a vessel or stream that is involved in the accident would be exhausted to the cell's ventilation system.

* A fire would release smoke contaminated with about 1% of the fission product and transuranic radionuclides contained in the burning organic solvent.

The above assumption is based upon measurements made at the Savannah River Plant, and is conservative relevant to actual experience with disposal of spent organic solvent by burning in open pits.

* The final filters would retain their integrity and efficency during and after the accident. Krypton, zenon, iodine and volatilized ruthenium radionuclides pass through the filters without being removed. The final filters would remove 99% of particulate in the form of smoke or aerosol.

This assumption is conservative based upon experience with absolute filters operating at their rated flow which have demonstrated better than 99.95% efficiency for removing particles greater than 0.3μ. It also has been demonstrated that these filters are approximately 99.5%

efficient for removing smoke ranging in particle size from 0.004μ to 0.03μ. The final filters are fire resistant and are in a protected location which is an adequate distance from the process cells so that they are not likely to be affected by the blast of explosion nor by a fire of limited duration in a process cell.

* The accident occurs during meteorology conditions that could result in a puff release (of Q curies) travelling at a rate of one meter per second. On this basis, a person who happens to be 400 meters downwind from the stack could receive the maximum total integrated dose (TID max.) of 2.34×10^{-5} Q(Ci-sec/m^3).

  This assumption is conservative. In order for persons to receive the maximum exposure, they would have to be in the area of maximum radionuclide concentration at ground level during the entire time the accidental release occurred. The area of maximum ground level concentration is within the exclusion area of the site, which upon warning would be evacuated.

4.4 Risk Evaluation

In general, for all accidents postulated, with the exception of accidental nuclear criticality, the potential release of plutonium and/or americium-curium radionuclides constitute the predominant hazard.

4.4.1 Nuclear Excursions

We have postulated a criticality accident in the dissolver totaling 10^{19} fissions prior to sufficient liquid or chopped fuel being ejected from the dissolver to render the system subcritical.

Further, we have assumed that all radioactive gases are released into the process cell, thus bypassing the off-gas systems, and subsequently are released to the atmosphere through the stack.

In the event of such an accident, we calculate a TID max. whole body exposure of 0.19 rem and a maximum potential inhalation exposure of 1.85 rem to the thyroid. These maximum exposures would occur within the exclusion area, at a point about 400 meters downwind from the stack. Beyond the site boundary, in the Barnwell area, exposures to the public would be at least a factor of three lower than those stated above.

4.4.2 Fires

We have evaluated the potential risk of fire in the process system, and have selected two hypothetical accidents as representing the upper-limit accident involving fires in the BNFP. In the event of a failure in the process system, alarms would signal an abnormal situation and prompt action would be initiated by the operator to mitigate the hazard. Upon detection of a flame, the fire detectors would be expected to automatically respond with a blanket of high density foam to put out the fire within about 5 minutes. However, for our evaluations we have assumed that a fire burns for a total of thirty minutes and is extinguished by oxygen depletion caused by pluggage of the cell filter.

HAP Solvent Fire - The HAP stream is the organic solvent stream leaving the HA contactor. This stream contains the largest amount of

plutonium per unit of time in an organic stream during normal operation, and it is also the organic stream with the highest fission product content during the neptunium accumulation cycle. For this hypothetical accident, we postulate a break in the line and ignition of the organic solvent spilled into the pan-liner. The organic solvent burns at a rate of 1 inch of depth per hour.

In the event of such an accident, based upon the assumptions stated above and in 4.3, we calculate that about 1 Ci of plutonium might be released from the stack. (Other radionuclides, which also would be released with the plutonium, do not add significantly to the consequences relevant to the hazard of plutonium.) Such a release would be about 1.1% of the upper-limit guideline.

3B(C) Explosion and Fire - The 3B Stripper strips plutonium from the organic phase into the aqueous phase. Hydrogen gas is generated in the 3B Pu Stripper, but is purged to the VOG system by nitrogen gas which is monitored by flow rate instruments with alarms. The operating temperature in this system is below 50°C, which is far below the auto-ignition temperature for a hydrogen-air mixture. The electrical system is designed to prevent current leakage, i.e., explosion-proof.

For this hypothetical accident we postulate the accidental introduction of air or oxygen into the system, and failure of the electrical insulation to prevent current leakage, i.e., somehow the hydrogen is ignited and an explosion occurs. Further, we postulate

the explosion expels the entire contents of the 3B(C) via a rupture in the vessel, and that the 3AP organic solvent stream is ignited by the explosion. We also assume that for some unexplainable reason the organic solvent stream (3AX) and related process streams to the 3A column do not stop, and thus the 3AP organic solvent stream continues to flow after the accident. In addition, we have assumed a delayed response by the fire detectors in the process cell, which normally would be expected to automatically respond with a blanket of foam to extinguish such fires within about 5 minutes.

In the event of such an accident, based upon the assumptions stated above and in 4.3, we calculate that about 150 Ci of plutonium would be dispersed into the cell air by the explosion, and about 150 Ci of plutonium would be in the smoke going to the cell's filter. This could result in a release of about 3 Ci of plutonium from the stack. Such a release would be about 3.6% of the upper-limit guideline.

There are three plutonium stripping cells in the BNFP, such as the one evaluated above, where there is some potential for an explosion to occur. The 3B(C), however, represents the upper-limit accident potential since it is the unit that contains solutions with the highest concentration of plutonium.

4.4.3 Explosions

In addition to the three Pu stripping cells, there are two waste evaporators in the BNFP where there is some potential for nitrating organic solvent, which could form an unstable compound (red oil) that could explode under certain conditions.

In order for the unstable compound (red oil) to form in the evaporator, entrained organic solvent must accumulate and be retained in the evaporator. In continuous-type evaporators, such as used in the BNFP, the accumulation of organic solvent will be limited by decanters, steam distillation, and constant bleed-off of evaporator bottoms. Even if "red oil" accumulates, an explosion will not occur unless the temperature of the solution exceeds 135°C. Temperature and density controls are provided to assure that the solution temperature will not approach the 135°C limit.

Nevertheless, while it does not appear possible that an explosion could occur in this type of evaporator, we have evaluated the potential consequence resulting from an explosion in both the HAW concentrator for high-activity waste and the LAW concentrator for low-level radioactive waste.

HAW Concentrator - For this hypothetical accident, we have assumed an abnormal condition with a plutonium concentration in the HAW concentrator equivalent to a 5% product loss rate. Based on the latter assumption and the assumptions stated in 4.3, we calculated that an explosion in the HAW concentrator might potentially release from the stack: 2400 Ci Ru-106, 3.2 Ci Zr-95, 0.8 Ci Sr-89, 0.8 Ci Sr-90, 1.7 Ci Cs-134, 1.1 Ci Cs-137, 8.9 Ci Ce-144, 0.5 Ci Am-Cm, 0.08 Ci Pu, and other lesser radionuclides. Of these, the Ru-106 and Am-Cm radionuclides are the predominant hazard. The total release, however, is 1.8% of the upper-limit guideline.

LAW Concentrator - For this hypothetical accident, we have assumed a condition which caused the plutonium concentration in the LAW concentrator to be equivalent to a 10% recycle rate. Based upon the latter assumption and the assumptions stated in 4.3, we calculated that an explosion in the LAW concentrator might potentially release from the stack: 33 Ci Ru-106, 35 mCi Pu, and other lesser radionuclides. The total release, however, is 0.06% of the upper-limit guideline.

4.4.4 Other Incidents

In addition to evaluating postulated accidents, we have reviewed the various possible radiological incidents that could occur in the BNFP. Allied-Gulf has identified the various potential abnormal events which might occur in the plant and has identified the detection, prevention, correction or control, and general consequences of these abnormal events in Section X of the PSAR, in Tables X-3 through X-19, titled Abnormal Events. In general, during normal operation of the BNFP, potential incidents may occur that could result in a release of radioactivity to the atmosphere or to surface streams at instantaneous concentrations which temporarily could exceed the annual average concentration values in Table II, 10 CFR Part 20. Releases of radioactivity from such incidents, on an average annual basis, are not expected to result in significant contamination of the environment and would be within the limits of 10 CFR Part 20

4.5 Conclusion

In general, we believe the assumptions we have made to evaluate postulated accidents are conservative, and thus, exaggerate the potential consequences of the above postulated upper-limit accidents with respect to the BNFP. However, our accident analyses do indicate the relative magnitude of postulated accidents and the maximum potential risk to persons in the vicinity of the plant if such events should occur. The potential exposure to persons beyond the site boundary in the Barnwell area would be at least a factor of three lower than the potential exposures we have estimated for persons within the site boundary, because we evaluated the consequences of the above accidents at the point of maximum potential exposure, which is within the site boundary, and the exposure would be that much less at the site boundary due to atmospheric dispersion. The following is a summary of our independent accident evaluation, and Allied-Gulf's evaluation, expressed as percent of the upper-limit guideline for nuclear fuel processing plants, which is comparable to the Reactor Site Criteria in 10 CFR 100. The reasons for the difference in values are explained below.

Upper-Limit Accident	% Stack Release Criteria	
	AEC Evaluation	Allied-Gulf's Evaluation 1/
Solvent Fire - 3B(C)	3.6%	0.75% 2/
Explosion - HAW	1.8%	1.00% 3/
Solvent Fire - HAP	1.1%	0.40% 4/
Nuclear Excursion	0.76%	0.30% 5/
Explosion - LAW	0.06%	0.04% 6/

1/ To assess the significance of certain assumptions relevant to the accident analyses, we elected to use assumptions in our evaluation which affected the consequence adversely. We are in general agreement with Allied-Gulf's analyses, however, and based upon our independent evaluation we have concluded that these accident analyses indicate the degree of risk to persons in the vicinity of the plant if such accidents should occur.

2/ Solvent Fire - 3B(C) Allied-Gulf's evaluation is based on a response within 5 minutes by the fire protection system in the cell. The effect also is governed by the aerosol filtration efficiency that is assumed for the HEPA filters.

3/ Explosion - HAW Our evaluation is based upon an abnormal process condition rather than a normal process condition. This has the effect of increasing the amounts of transuranic nuclides that could be released as a result of such an explosion.

4/ Solvent Fire - HAP Allied-Gulf's evaluation is based upon a process condition which effects a maximum fission product release during a neptunium recovery campaign. Our evaluation is based upon normal process conditions and indicates the effects of a burning plutonium-bearing organic solvent stream.

5/ Nuclear Excursion Allied-Gulf postulated that this event would result in about 10^{18} fissions, and calculated the effect without decay time allowances. We postulated the event could yield ten times more fissions, but calculated the effect with decay time allowances.

6/ Explosion - HAW The difference reflects the aerosol filtration efficiencies that were assumed, i.e., 99% vs 99.86%.

5.0 RESEARCH AND DEVELOPMENT

The following development projects are underway or planned by Allied-Gulf to improve demonstrated technology, to demonstrate new technology, and to verify equipment design for economic and safety benefits. We have concluded that there is reasonable assurance that these projects will be completed prior to the need for application of the technology in the BNFP.

5.1 Plutonium Partition and Stripping

Pilot plant tests of the new process and related equipment are in progress to demonstrate the reliability of this new technology for plutonium partition and stripping operations. The results of these tests will be reviewed by us and are subject to our approval before this technology is used in the BNFP. Acceptable alternatives, based upon demonstrated technology, have been identified in the event these tests are not successful.

5.2 Fuel Bundle Shear

Development of the design and tests of a fuel bundle shear and the seals between the shear and the dissolver are in progress to verify the adequacy of the equipment.

5.3 Iodine Scrubber

The mercurous-mercuric nitrate-nitric acid scrubber system for removal of iodine is being studied to determine whether high concentrations of nitrogen oxides at high temperatures will have any effect on the iodine removal efficiency.

5.4 Denitration

Alternative methods for the destruction of nitric acid with sugar, such as denitration in combination with concentration in the evaporator, are being studied to reduce ruthenium volitalization.

5.5 Fissile Material Detectors

Remotely-operated fissile material detectors are being developed for safeguarding special nuclear material.

6.0 QUALITY ASSURANCE PROGRAM

Allied-Gulf has developed a plan for a comprehensive quality assurance program. The scope of the plan extends from detailed design of the facility and will continue during operation of the plant. The objective of this program is to assure that the plant will be constructed and maintained to meet the safety reliability objectives identified in the Final Safety Analysis Report.

The project staff consists of technical personnel experienced in design, construction, and operation of chemical processing plants, who will review and approve all work performed by its contractors. Allied-Gulf will directly participate in the qualification and selection of vendors, fabricators, and other sub-contractors. Although inspection of the equipment during fabrication and installation will be performed by the Bechtel Corporation, Allied-Gulf will approve their quality control programs and in addition, Allied-Gulf will perform, throughout construction, independent inspections and audits to ensure quality of components critical to safety.

Overall, Allied-Gulf acknowledges their corporate and public responsibility relevant to design, construction, and safe operation of this plant.

Allied-Gulf has scheduled a one-year period for post construction cold start-up operations to verify the performance capabilities and protection systems of the BNFP, and to complete operator training.

We have concluded that Allied-Gulf's quality assurance program is in accord with Appendix B to 10 CFR Part 50 and is acceptable.

7.0 ORGANIZATION AND TECHNICAL QUALIFICATIONS

Among the three applicants, Allied Chemical Nuclear Products, Inc., has the primary responsibility for the design, construction, and operation of the plant. The Bechtel Corporation will provide the architect-engineering and engineering-construction services. Personnel assigned the responsibility for this project have extensive experience in the design, construction, and operation of spent reactor fuel reprocessing plants.

The Project Group, which is identified in Exhibit E of the application (See Amendment No. 5), will be restructured upon completion of the BNFP and it also will be the operating organization. That is, the persons who have responsibility for design and construction of the BNFP, will have responsibility also in related functional components of the organization established for plant operation. This organizational approach capitalizes on knowledge obtained during the project phase, and assures that the management and supervisory personnel will be familiar with the facility with respect to its capabilities, limitations, and safety considerations.

Plant operators and maintenance personnel will receive classroom and on-the-job training to prepare them for their assignments. This will be done during post construction check-out and preoperation testing of the BNFP. The curriculum for the training program will be developed as design and construction proceeds. Plant operations personnel who must be relied upon to exercise appropriate judgments regarding the operation of the process will be subject to operator license requirements of 10 CFR Part 55.

Prior to "hot" operation, technical specifications will be developed for our approval, which will define the restraints and limits to be imposed by the operating license. In addition, detailed instructions (Standard Operating Procedures) will be written for the operation of all components of the BNFP. These will include instructions for start-up, routine operation, permissible range of operating conditions, handling emergencies and shutdown procedures.

Three separate committees will be established to (a) approve and maintain current the Standard Operating Procedures and to recommend changes in technical specifications, (b) review and approve changes in Standard Operating Procedures or technical specifications related to nuclear criticality considerations, and (c) review and approve changes affecting the accountability of special nuclear material.

The Bechtel Corporation's experience in the nuclear field dates back to 1949. Bechtel was responsible for the construction of the AEC's Idaho Chemical Processing Plant, the design and construction of the Nuclear Fuel Services, Inc., Spent Fuel Reprocessing Plant, and Hot Cell Facilities for General Electric, for Atomics International and for General Atomic. In addition, Bechtel has designed and constructed other nuclear facilities including power reactors.

We have concluded that Allied-Gulf and their contractors are qualified to design and construct the Barnwell Nuclear Fuel Plant.

8.0 EMERGENCY PLANNING

Allied-Gulf has defined the objectives and the scope of an emergency plan. The emergency plan will include step-by-step procedures, and delineation of authority and responsibility for implementing action. Emergency planning will be coordinated with all necessary local, state and Federal agencies. An agreement will be made with a hospital for equipment and space to receive contaminated injured people.

The services external to the site are well developed considering the remoteness of the site from population centers. This is largely due to the long-term influence of the SRP in developing a nuclear preparedness posture in the local public service organizations.

We have concluded that the objectives and the scope for the emergency plan are acceptable. The detailed plan will be reviewed at the operating license stage.

9.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) has reviewed Allied-Gulf's application for a permit to construct the Barnwell Nuclear Fuel Plant. The ACRS completed its review of the proposed facility at its 123rd meeting, on July 10, 1970. A copy of the ACRS report to the Commission, dated July 17, 1970, is attached as Appendix A.

In the report, the ACRS made several recommendations which are discussed below.

9.1 Integrity of the Waste Storage System

Allied-Gulf will give careful attention to the design, fabrication and installation of the high-activity liquid waste storage tanks, the waste vault liners, and related confinement systems.

The ACRS recommended that Allied-Gulf give consideration also during the design of the waste tank vault liners to possible means for assuring the continued integrity of the vault liners and to provide adequate margins against corrosion in the waste confinement systems. Allied-Gulf is performing corrosion studies using simulated solutions which have compositions similar to the acid waste solutions anticipated from the BNFP process. The results of corrosion tests, to qualify the proposed materials to be used in the high-activity waste storage system, will be submitted to us for our review. Corrosion studies will be continued during actual operation of the waste storage system. Allied-Gulf's quality assurance program and procedures will be reviewed by the AEC to assure that careful attention is given to the design, fabrication and installation of the high-activity waste storage system.

9.2 Fuel Pool

The ACRS recommended that Allied-Gulf give further consideration to whether an atmospheric cleanup system should be provided in the Fuel Receiving and Storage Station to cope with potential gaseous releases in the event fuel elements were damaged during cask unloading operations. Allied-Gulf will perform such studies and will evaluate whether suitable filters can be installed to mitigate iodine releases to the atmosphere in the event that fuel elements are damaged during unloading operations.

Allied-Gulf will design the stack to be a Class I structure and it will be located at a distance that will prevent possible damage to the fuel pool if the stack should fail.

9.3 Filter Integrity

The ACRS recommended that during detailed design attention should be given to assurance of filter integrity during blast loading. At the review for an operating license, Allied-Gulf will be required to show that their detailed design provides protection of the filters from possible blast loadings.

9.4 Conclusion

The ACRS concluded in its letter that it believed that the items mentioned above can be resolved during construction, and that, if due consideration is given to the foregoing, the Barnwell Nuclear Fuel Plant can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

10.0 COMMON DEFENSE AND SECURITY

The activities to be conducted by Allied-Gulf will be within the jurisdiction of the United States. Allied-Gulf Nuclear Services is a partnership of Allied Chemical Nuclear Products, Inc., and Gulf Energy & Environmental Systems, Inc., both of which are Delaware corporations. All of the directors and principal officers of these two corporations are United States citizens. Allied-Gulf is not owned, dominated nor controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but Allied-Gulf has agreed to safeguard any such data which might become involved in accordance with the regulations. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

11.0 FINANCIAL QUALIFICATIONS

We have reviewed the financial information presented in the application and amendments thereto. Allied Chemical Corporation and Gulf Oil Corporation have each guaranteed performance and assumed each and every liability assumed by their respective wholly-owned subsidiary under the partnership agreement dated February 1, 1970. Accordingly, in the final analysis, the applicants Allied Chemical Nuclear Products, Inc., and Gulf Energy & Environmental Systems, Inc., for the funds needed by each to meet design, construction and plant startup costs, will rely on the resources, including operating revenues, of Allied Chemical Corporation and Gulf Oil Corporation respectively, when and as required. Information contained in the application, as amended, indicates that such resources will be ample to cover the estimated cost of BNFP. Therefore, we have concluded that the applicants are financially qualified to engage in the design and construction of the proposed facility. Our detailed evaluation of financial qualifications is attached as Appendix F.

12.0 CONCLUSIONS

Based on the proposed design of the Barnwell Nuclear Fuel Plant, on the criteria, principles, and design arrangements for the systems and components thus far described, which include all of the important safety items, and on the calculated potential consequences of routine and accidental release of radioactive materials to the environs, on the scope of the development program which will be conducted, and on the technical competence of Allied-Gulf and the principal contractors, we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR Part 50 and paragraph 2.104(b), 10 CFR Part 2:

1. Allied-Gulf has described the proposed design of the facility including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public;
2. Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration will be supplied in the final safety analysis report;
3. Safety features or components, if any, which require research and development have been described by Allied-Gulf and Allied-Gulf has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components;

4. On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

5. Allied-Gulf is technically qualified to design and construct the proposed facility;

6. Allied-Gulf is financially qualified to design and construct the proposed facility; and

7. The issuance of a permit for the construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 17, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON THE BARNWELL NUCLEAR FUEL PLANT

Dear Dr. Seaborg:

At its 123rd meeting, July 9-11, 1970, the Advisory Committee on Reactor Safeguards reviewed the proposal by Allied-Gulf Nuclear Services to build the Barnwell Nuclear Fuel Plant at a site about seven miles west of Barnwell, in Barnwell County, South Carolina. The site is contiguous with the eastern boundary of the Savannah River Plant. The project has been considered previously at Subcommittee meetings on October 2, 1969 at the site and on June 3, 1970, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives of Allied-Gulf Nuclear Services, and the AEC Regulatory Staff and its consultants, and the documents listed.

The plant will be designed to recover 1,500 metric tons of uranium per year with a daily capacity of 5 metric tons. In the process, fuel bundles are sheared into short lengths and fed to a nitric acid dissolver where uranium, plutonium, neptunium, and most fission products are dissolved. The solution is separated from the cladding and processed further. Among the major steps is recovery of uranium and plutonium as nitrates by solvent extraction of their nitric acid solutions with tributyl phosphate.

The feasibility of the various basic operations to be employed in this plant has been demonstrated. However, Allied-Gulf Nuclear Services is continuing work needed to verify detailed process characteristics where appropriate. The Committee wishes to be kept informed. The start-up schedule allows for a one-year period of cold operation to verify performance characteristics and to complete operator training.

High activity liquid waste will be stored as a nonboiling nitric acid solution for future solidification and transfer to a Federal repository. Solid wastes with known or detectable transuranic nuclides will be segregated, packaged and buried in identified locations so as to assure that they can be retrieved.

Careful attention should be given to design, fabrication and installation of the high level liquid waste tanks and the waste tank vault liners. Consideration should be given to possible means for assuring the continued integrity of the vault liners and to the provision of adequate margins against corrosion.

Present plans do not call for exhaust filtration of the fuel pool atmosphere. Should further studies show the need for fuel pool atmosphere cleanup, suitable filters will be installed. The applicant stated further that, should the pool prove vulnerable to a failure of the ventilation system exhaust stack, the stack will be installed at a distance that will prevent damage to the pool.

The applicant stated that the normal power system, the emergency power system, and related equipment will be designed so that no single failure will interrupt operation of the ventilation system or other vital services.

Attention should be given to assurance of filter integrity during blast loading. Provisions will be made for shutting the plant down safely from outside the control room should the control room become uninhabitable.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved during construction, and that, if due consideration is given to the foregoing, the Barnwell Nuclear Fuel Plant can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Chairman

Attachment
List of Documents

References - Barnwell Nuclear Fuel Plant

1) Application for Construction Permit and Operating License

2) Vols. I and II, Barnwell Nuclear Fuel Plant Safety Analysis Report

3) Proprietary Supplement to Safety Analysis Report

4) Amendments Nos. 1, 2, 3, and 4

5) Addenda Nos. 1, 2, 3, 4, 5, and 6

APPENDIX B

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

JUN 22 1970

Mr. J. A. McBride, Director
Division of Materials Licensing
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. McBride:

Transmitted herewith in response to your request of December 4, 1968, is a review of the geologic and hydrologic aspects of Barnwell Nuclear Fuel Plant AEC Docket No. 50-332 - proposed by the Allied-Gulf Nuclear Services.

This review was prepared by H. H. Waldron, P. J. Carpenter, R. Schneider, and D. G. Metzger and has been discussed with members of your staff. We have no objections to your making this review a part of the public record.

Sincerely yours,

W A Radlinski

Acting Director

Enclosure

cc: Mr. Walter G. Belter, AEC

Allied-Gulf Nuclear Services
Barnwell Nuclear Fuel Plant

AEC Docket 50-332

The planned location of the Barnwell Nuclear Fuel Plant is in Barnwell County, South Carolina, immediately inside (west of) the present eastern boundary of the Savannah River Plant and is approximately 1½ miles east of Par Pond (an impoundment across Lower Three Runs Creek). It is 7 miles east of Barnwell and 31 miles southeast of Augusta, Georgia. The purpose of the plant will be to recover fissionable materials from irradiated nuclear power reactor fuel elements which, prior to irradiation, consisted of uranium oxide or a mixture of plutonium and uranium oxide clad in stainless steel or zirconium alloys. Plant products will be nitrate solutions of uranium, plutonium, and neptunium. The plant will use a purex type aqueous process similar to those used at the Atomic Energy Commission plants at Hanford, Savannah River, the National Reactor Testing Station, and the West Valley Plant of Nuclear Fuel Services, Inc. Nominal capacity of the plant will be 5 metric tons of uranium per day.

The analyses of the geology and hydrology of the site as presented in the "Safety Analysis Report" (SAR) and supplements, were reviewed and compared with the available data and literature and with similar data from the Savannah River Plant area. Geological conditions at the site were examined on May 22, 1969. Except as noted below, the analyses presented appear to be an adequate appraisal of those aspects of the hydrology and geology that are pertinent to an engineering evaluation of the adequacy of the site.

Geology

The site is located near the eastern edge of the Aiken Plateau portion of the Atlantic Coastal Plain physiographic province. In the site area about 800 to 900 feet of nearly flat-lying, unconsolidated to poorly consolidated, Upper Cretaceous to Quaternary sediments unconformably overlie older, well consolidated rocks. These older rocks are presumed to be a continuation of those that crop out some 30 miles or more to the west in the Piedmont province. At the site they include Triassic sedimentary rocks and some crystalline rocks of Precambrian or early Paleozoic age. Based on geophysical data, the Triassic rocks appear to occupy an elongated northeast-trending basin that underlies the site. Commonly such buried Triassic basins are thought to be bounded on one or both sides by northeast-trending faults; however, any tectonic activity associated with the basins does not appear to have involved near-surface differential displacements since late Triassic time.

According to the SAR the stratigraphic sequence at the plant site includes a surficial eolian sand, a few feet thick, that is underlain by 14 to 30 feet of sediments of the Miocene Hawthorn Formation, chiefly interbedded sandy clay and clayey fine sand. Underlying the Hawthorn is about 38 to 59 feet of sediments of the late Eocene Barnwell Formation, chiefly beds of silty, fine to medium sand and some clayey sand; a persistent kaolinitic clay layer occurs in the upper part of the formation. Approximately 48 to 114 feet of sediments of the early Eocene McBean Formation, chiefly beds of silty, fine to medium sand, underlies the Barnwell Formation.

Although numerous, shallow surface depressions, or sinks, occur throughout this part of South Carolina, and several also occur in the vicinity of the site, none are present at the actual plant site. Nor is there any evidence from the subsurface investigations that the zone of marl and calcareous sand, which has been reported to occur in the middle part of the McBean Formation in other localities in South Carolina, is present at the plant site. Consequently, no need for remedial treatment of the foundation for solution phenomena is anticipated.

The applicant proposes to place a 15-foot-thick, pre-construction surcharge fill over the plant site. Then, several months later, this fill will be excavated to the desired foundation grades for the major plant structures, most of which will be founded in the underlying clayey sand and sandy clays of the Hawthorn Formation or in the upper part of the underlying Barnwell Formation.

Although there are no identifiable faults other young geologic structures in the area that might be expected to localize seismicity in the immediate vicinity of the site, structural details in the older rocks underlying the Coastal Plain are poorly known, and epicenters of earthquakes that have occurred in the area normally cannot be related directly to any specific geologic structure. In other parts of eastern United States, however, some low-level seismicity appears to be related to the Triassic basin border faults. Available data suggest that the regional structural trends in these older rocks are mostly northeastward, paralleling those in the adjoining Piedmont province to the west. Locally, however, younger, broad regional structures that trend northwestward are superimposed on the older northeastward trends. The numerous earthquakes that have occurred in the vicinity of Charleston, for example, appear to be localized along one of these northwest-trending structures.

Hydrology

Surface drainage from the site, which is situated on a local topographic divide, will be eastward toward Salkehatchie River and south and westward toward Lower Three Runs Creek. As there are no streams traversing the site and the elevation of the plant and the grade of the proposed burial ground are 40 feet above that of the roadway across Par Pond, flooding at the site would be unlikely.

Regarding the release of radioactive materials with the liquid effluent to Lower Three Runs Creek, the applicant has stated that during normal operation of the plant no releases are planned, that abnormal (non-accident) releases will be controlled to the lowest practical level below 10 CFR 20 limits, and that accidental releases will not result in offsite exposures exceeding those specified in 10 CFR 100. Lower Three Runs Creek, which flows southward 14 miles where it joins the Savannah River, is inside the boundary of the Savannah River Plant. Based on some 63 years of streamflow record at Augusta, Georgia, the observed average and minimum daily discharges of the Savannah River at its junction with Lower Three Runs Creek are about 10,000 and 1,000 cubic feet per second respectively. Abnormal or accidental releases of radioactive effluents which are within the release limits should be well diluted and dispersed with Savannah River water prior to its use for domestic purposes some 90 miles downstream. If releases of radioactive materials to Lower Three Runs Creek were possible in quantities greater than those allowed under the release criteria, an analysis should be made of their movement and (or) retention in the hydrologic environment, including an evaluation of the level of radioactivity to which the public might be exposed.

Cooling water, which will be required on a continuous basis for the operation and safe shutdown of the plant (2,400 and 1,200 gallons per minute, respectively), will be supplied by four wells completed in the Tuscaloosa Formation of Late Cretaceous age. The applicant states that two of the wells can supply the amount of cooling water required for the safety of the plant. Based on the known hydrologic characteristics of the Tuscaloosa aquifer (Siple, 1967)1/, an adequate supply of cooling water should be assured if an event which could cause failure of two wells would not cause failure of all four wells.

In general, the matter of most serious concern at this site is the possible movement of radionuclides into the ground-water supplies of the surrounding area. The applicant states that there are possibilities of leakage of radionuclides from the fuel pools and of leaching of radionuclides from the burial sites, and that the potential consequences of three abnormal events (specified in table X-17) is the release of radioactive liquids to the roadways and offsite. However, it is also stated that under these abnormal conditions the concentration of radioactive liquids released to unrestricted areas will not exceed 10 CFR 20 concentration limits. The quantities and types of radionuclides which could be released at the site and possibly could migrate to the ground-water reservoir are not given in the application; hence, it would be impossible for us to verify the applicant's statements concerning their concentrations in unrestricted areas.

1/ Siple, G. E., 1967, Geology and ground-water of the Savannah River plant and vicinity, South Carolina: U.S. Geol. Survey Water-Supply Paper 1841.

Regarding the disposal of solid waste at the site, the applicant states (p. II. 5-3) that these wastes will be buried above the water table. Before the solid waste disposal facility is designed, consideration should be given to the supplemental water-table information to be obtained in connection with the recommendations of the next paragraph. The applicant also states that mounding of earth on top of the filled trenches of the burial ground, together with ditching and diking, will minimize percolation of rainwater through the buried wastes. Details of the procedures to be used are not given. It appears that such practices could concentrate the runoff between trenches and thereby increase the amount of percolation through the buried wastes. The extent of percolation of rainwater to the buried wastes will depend on the permeability of the mounded earth as well as the effectiveness of the ditching and diking practices to be used.

The applicant has computed the time required for contaminated liquids spilled at the plant grade to move 50 feet downward to the water table and then to Lower Three Runs Creek, to be 2,200 years; 500 years vertically and 1,700 years horizontally. Although they appear to be of a reasonable order of magnitude, these travel times are not entirely reliable because all the computations ignore the porosity, which would shorten the travel times considerably, and because the depth to water used in the computations may be incorrect. In addition, as noted above, the waste radionuclides may be released from the bottoms of the fuel pools or waste burial trenches as well as from the plant grade. The applicant has prepared a ground-water contour map in which the depths to water have been interpreted from a seismic refraction survey. The seismic refraction survey consists of measuring changes in the velocity of a seismic wave with depth. The seismic wave is produced by detonating an explosive and the times of arrival are recorded at an array of geophones. The depth to ground water interpreted from such velocity changes may or may not be correct. For a ground-water contour map of this type to be reliable, it should be based primarily on a sufficient number of depth-to-water measurements in cased holes. Many of the water levels measured by the applicant are not consistent with the map prepared from the geophysical data; some are higher, suggesting either the presence of a perched water body, or a more shallow water table. In addition, there may be large seasonal water-table fluctuations. For example, Siple (1967, p. 77) shows seasonal fluctuations as large as 20 feet in wells tapping a shallow water-table aquifer near the center of the Savannah River Plant site; also, periodically the water table rises nearly to land surface. Another factor that should be considered in selecting a reasonable and proper depth to the water table for use in computing the travel time of ground water, is related to long-term fluctuations of the water table. These fluctuations would result from protracted wet or dry periods of several years duration. The magnitude of these fluctuations should be estimated from available water-level data in shallow wells in the same general area and from an analysis of precipitation records. It would appear, therefore, that the depth to water that should be used in computing vertical travel times may be as little as a few feet rather than the 50 feet used by the applicant.

The applicant concludes, on the basis of his ground-water contour map and the surface topography between drill holes 2 and 22, that the water-surface contours conform generally to those of the surface topography and that the flow of ground water from the site would be to the south and west, or to the east and then south to Lower Three Runs Creek. This conclusion is not supported by a comparison of the surface topography and ground-water contours near other drill holes such as 1 and 3. These data suggest that some ground-water contours may not conform to the surface topography. It is possible, therefore, that some ground-water movement may be eastward, offsite, toward the Salkehatchie River.

To assess the ground-water travel times from points of possible release of radionuclides at the plant to unrestricted areas, a water-table map of the shallowest aquifer, based on water-level measurements, should be provided. The map should encompass an area large enough to evaluate the possibility of eastward movement of ground water toward the Salkehatchie River as well as the southwestward movement to Lower Three Runs Creek. It should be detailed enough to evaluate the hydraulic gradients in all parts of the area. The travel times should be reevaluated based on the highest estimated position of the shallowest water table, using the porosity values in the computations, and the hydraulic gradients from the revised water-table map.

The applicant concludes that radioactive materials which would enter the shallow water-table aquifer would not migrate downward to the confined aquifer because of the low permeability of the material separating the two, and because of the moderate pressure in the confined aquifer. This interpretation is contradicted by the information obtained by Siple (1967) who demonstrated that recharge to the Tuscaloosa and overlying aquifers, which are confined, is primarily from surface infiltration. The data given in the SAR indicate that some of the water levels in the Barnwell are below and some are above the water level in the shallow water-table aquifer. Siple (1967, p. 78) presented further evidence of a hydraulic connection between the water-table and the confined aquifers by demonstrating that pumpage from the Tuscaloosa is reflected in water-level changes in the shallower aquifers. If pumpage from the Tuscaloosa Formation were to increase significantly east of the site, radionuclides contained in the overlying aquifers could be expected to move downward and, depending on future conditions, possibly move into the Tuscaloosa and offsite. Several observation wells should be installed in the Tuscaloosa aquifer and in the shallow water-table aquifer to determine and monitor the head relationship between the two. If future pumpage from the Tuscaloosa results in an excessive lowering of head, the rate of movement of ground water between the two should be evaluated by a pumping test.

The SAR states that wells in the vicinity of the solid waste burial area and the processing facility will be monitored to determine if radionuclides have migrated to the ground water. In addition, it is stated that if the ground water were to receive radionuclides, the paths they would follow would be closely observed by wells around the exclusion area; and action would be

taken, as necessary, to keep the level of radioactivity in the offsite environment within acceptable limits. However, the number and spacing of monitoring wells are not given and the course of action to be taken to minimize the activity level beyond the exclusion area is not described.

Although monitoring wells are of value at the site of nuclear facilities, it must be remembered that the data obtained from the monitoring will not necessarily prove that radionuclides are not migrating from the site. In other words, the absence of radionuclides in samples obtained from a monitoring system does not prove containment of radionuclides on the site. Because of the possible complexity in the flow pattern of ground water, radionuclides contained in it could bypass the monitoring wells and not be detected until they have moved some distance from the site. If this were to happen, it might be impossible or impracticable to take effective remedial action.

In summary, the applicant has not shown that radionuclides buried at this site or leaked from the plant could not migrate downward to the aquifers. For the following reasons the rates of ground-water movement, as computed by the applicant, may not be entirely valid: a) the water table may, at times, be higher than that estimated by the applicant, b) the vertical distances from possible sources of radionuclides to the water table may be much less than those shown, c) the hydraulic gradients may be greater than those estimated, and d) the porosity values were not used in the computations. Also, it should be emphasized that present hydrologic conditions will undoubtedly be changed by future ground-water developments which will lower water levels, induce more rapid rates of flow toward the wells, and alter the direction of ground-water movement.

If one assumes that the computed rates of ground-water movement are of the right general order of magnitude, there would appear to be little cause for concern about the short-lived radionuclides moving into nearby water supplies. Although the order of magnitude appears to be reasonable, the applicant should verify his calculations by obtaining the additional data suggested above and considering the above-stated reasons for doubting their validity.

The matter of greatest concern from a long-range standpoint is the possibility and consequences of long-lived radionuclides, particularly plutonium, moving undetected into the hydrologic environment. It is doubtful that one could demonstrate convincingly that such radionuclides could be effectively isolated and immobilized for tens or hundreds of thousands of years under the hydrologic and geologic conditions of the site. An example of one process by which these radionuclides conceivably could be released to surface streams, long after burial or seepage underground, is erosion of the site. In view of the fact that the solubility and mobility characteristics of plutonium are poorly understood, it appears that the possibility of it moving with the ground water should not be ruled out. Therefore, any assurance of isolating and immobilizing such wastes for an indefinite period of time should be viewed with caution.

APPENDIX C

U.S. DEPARTMENT OF COMMERCE
Environmental Science Services Administration
COAST AND GEODETIC SURVEY
Rockville, Md. 20852

Reply to
Attn of: C23

MAY 7 1970

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of Barnwell County, South Carolina. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismic activity of the area as presented by the Allied Chemical Corporation in the "Preliminary Safety Analysis Report," for use in the evaluation of the site of the proposed Barnwell Nuclear Fuel Plant; and we hereby submit our conclusions concerning the seismicity factors.

If we may be of further assistance to you, please contact us.

Sincerely,

for Don A. Jones
Rear Admiral, USESSA
Director, C&GS

10 Enclosures

REPORT ON THE SITE SEISMICITY FOR

THE BARNWELL NUCLEAR FUEL PLANT, SOUTH CAROLINA

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed Barnwell Nuclear Fuel Plant near Barnwell, South Carolina, and has reviewed a similar analysis presented by the applicant in the "Preliminary Safety Analysis Report." The applicant's report on the site seismicity is comprehensive and adequate for the determination of the seismic factors for this site.

The most predominant seismic event to affect this site was the Charleston, South Carolina earthquake of August 31, 1886. This event, located approximately 80 miles southeast of the proposed facility, produced an estimated intensity of VI (MM) at the site. It is assumed that an earthquake with an intensity equal to or slightly higher than the 1886 earthquake could again occur in the same vicinity as the previous event.

This proposed facility is located on approximately 750-1200 feet of coastal plain sediments with no identifiable geologic or tectonic structure that would localize earthquakes in the vicinity of the site. Therefore, it must be assumed

that moderate earthquakes, intensity VI (MM), similar to those which have occurred in the Piedmont and Coastal provinces could occur in the general vicinity of the site.

As a result of this review of the seismological and geological characteristics of the proposed plant site area, and on the assumption that an adequate and stable foundation is provided, the Coast and Geodetic Survey agrees with the applicant that an acceleration of 0.12 g, resulting from an intensity VI - low VII (MM) earthquake, is adequate for representing earthquake disturbances likely to occur within the lifetime of the facility. The Survey also agrees that an acceleration of 0.20 g, resulting from an intensity VII or low VIII (MM) earthquake, is adequate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would provide an adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

May 5, 1970

Comments on
Barnwell Nuclear Fuel Plant
Allied Chemical Nuclear Products, Inc.
Safety Analysis Report
Volumes I and II dated November 7, 1968

Prepared by
Air Resources Environmental Laboratory
Environmental Science Services Administration
January 10, 1969

The site and its environs is a generally flat and heavily wooded region, and should present no unusual meteorological factors with regard to atmospheric transport and diffusion. A comprehensive diffusion climatology has been assembled from instrumentation on a 1200-ft TV tower located 25 miles to the northwest of the site. It is our opinion, that these data are applicable to the site because of the relatively short distance between the site and the tower, and the similarity of terrain and vegetation.

From the data presented in Appendix H, it can be shown that on an annual basis the prevailing flow is from the southwest at a frequency of 9 percent. This is divided among three diffusion regimes, namely, 45 percent unstable, 10 percent neutral, and 45 percent stable with wind speeds averaging about 5 m/s. On this basis, assuming a 100 m emission height, we compute the maximum average annual concentration to be 1×10^{-7} sec m^{-3} at a distance of about 1 km. This agrees closely with the applicant's estimate as shown in fig. X-1.

For the upper limit accident case the applicant has used a range of diffusion rates from very unstable, 2 m/s to neutral, 5 m/s to moderately stable, 2 m/s. One might argue that a 1 m/s wind speed should be used in the unstable and stable cases since these conditions each occur about 1 percent of the time. However, this would tend to increase the doses listed in Table III of the Safety Analysis only by a factor of 2.

In summary, we are in general agreement with the applicant's analysis. The maximum annual concentration from a routine effluent released at a height of 100 m is about 1×10^{-7} uCi/cc per Ci/sec release and is at a distance of about 1000 m from the stack. At the nearest offsite boundary (2000 m to the east), the maximum ground level air concentration from an accidental release will be about 1×10^{-5} uCi/cc per Ci/sec released assuming Pasquill Type C diffusion and a 1 m/s wind speed.

APPENDIX D

JOHN A. BLUME & ASSOCIATES, ENGINEERS
612 HOWARD STREET • SAN FRANCISCO, CALIFORNIA 94105 • (415) 397-2525

May 4, 1970

Mr. Edson G. Case, Director
Division of Reactor Standards
U. S. Atomic Energy Commission
Washington D.C. 20343

Contract No: AT(49-5)-3011
Blume Project No: 2085503
Subject: Barnwell Nuclear Fuel Plant
Allied-Gulf Nuclear Services
Docket No. 50-332

Dear Mr. Case:

In accordance with your request, we have performed a general review of the Safety Analysis Report, Amendments, and Addenda for the Barnwell Plant. Our review was also based on data gained during meetings with members of the staffs of the Divisions of Material Licensing and Reactor Licensing and the applicant on July 28, 1969 and December 18, 1969, and during a site visit on May 22,1969.

As requested by Mr. Workinger we are enclosing herewith five copies of our final report, "Review of the Seismic Design Criteria for the Barnwell Nuclear Fuel Plant." This report is essentially the same as our April 10, 1970 preliminary draft. Our review has been primarily confined to the engineering factors pertinent to the seismic design criteria and seismic analysis procedures.

Please note that it is our opinion that the response spectrum proposed for the site is not adequate. We understand that the applicant intends to modify the proposed site spectrum in Addendum No. 4. However, we have not yet received this addendum for review.

Very truly yours,

JOHN A. BLUME & ASSOCIATES, ENGINEERS

Roland L. Sharpe
Roland L. Sharpe
Executive Vice President

RLS:aa
Enclosure: 5

REVIEW OF THE SEISMIC DESIGN CRITERIA

FOR THE

BARNWELL NUCLEAR FUEL PLANT

(AEC Docket No. 50-332)

May 4, 1970

JOHN A. BLUME & ASSOCIATES, ENGINEERS
San Francisco, California

REVIEW OF THE SEISMIC DESIGN CRITERIA FOR THE BARNWELL NUCLEAR FUEL PLANT (AEC Docket No. 50-332)

INTRODUCTION

This report summarizes our review of the engineering factors pertinent to the seismic design criteria for the Barnwell Nuclear Fuel Plant. The plant is located in Barnwell County, South Carolina, approximately seven miles west of the town of Barnwell and contiguous with the eastern boundary of the AEC Savannah River Plant. The Barnwell Plant will process irradiated nuclear power reactor fuel elements consisting of uranium oxide or a mixture of plutonium oxide and uranium oxide clad in stainless steel or zirconium alloys. The design and construction of the plant will be performed by Bechtel Corporation under the direction of the applicant, Allied-Gulf Nuclear Services. Application for a construction permit has been made to the Atomic Energy Commission (AEC Docket No. 50-332) by Allied-Gulf Nuclear Services. A Safety Analysis Report has been submitted in support of the application to show that the plant will be designed and constructed in a manner which will provide for safe and reliable operation. Our review is based upon the information presented in the Safety Analysis Report and is confined primarily to an evaluation of the seismic design criteria and seismic analysis procedures for Class I structures, systems, and components. The list of reference documents upon which this report is based is given at the end of this report.

DESCRIPTION OF FACILITY

The proposed site is on nearly flat lands which are part of a once-continuous plain that is now dissected by southerly draining stream systems. Three Runs Creek, which drains the region, has incised its channel to a depth of about 150 ft. below the nearly level surface of the plain near the site. The uppermost geologic formations are poorly consolidated quartzose sands of Tertiary age which are locally mixed with clay or contain clay interbeds. These formations are blanketed at the surface by residual soil

of clayey sand and by wind blown sand. The relatively undisturbed Tertiary formations extend to depths of about 160 ft. They are underlain by similar Cretaceous formations to a total depth of about 800 ft. Triassic rocks probably underlie the Cretaceous, and if this is so, it indicates that a Triassic structural trough underlies the area. Elsewhere the Cretaceous formations rest directly upon the Paleozoic-Cambrian crystalline basement.

The Process Building will consist of a complex of cells containing equipment which processes material containing gross amounts of radioactivity. The cells will be shielded from the surrounding aisles and support facilities by reinforced concrete confinement shielding. Reinforced concrete construction will also be used for the control room, storage basins, waste vault, decontamination facilities, and scrap removal areas. Process support and personnel operating areas will consist of various types of concrete and steel construction. The structures will be founded on conventional spread or mat footings.

STRUCTURAL DESIGN CRITERIA AND LOADS

All structures, equipment, systems, and piping are classified according to their function or the consequences of their failure as Class 1, 2, or 3 as defined in Section V of the Safety Analysis Report. Class 1 structures, systems, and equipment are those whose failure could cause uncontrolled release of radioactivity, or those whose function is required to effect and maintain a safe plant shutdown. Class 2 structures and systems are those whose failure would not result in an uncontrolled release of radioactivity and whose function is not required to effect and maintain a safe plant shutdown. Class 3 structures, and systems are those whose complete failure could inconvenience plant operation, but which are not essential for safe operation, shutdown, or maintenance of the plant.

The design loads for the Barnwell Nuclear Fuel Plant are divided into two basic categories. The first category includes normal operational loads (dead, live, pressure, thermal, and dynamic) and the second includes abnormal loads (earthquake, wind, tornado. and missile impact).

Class 1 and 2 structures will be designed so that maximum stresses resulting from the design loadings will not exceed allowable working stresses as given in the Southern Standard Building Code or the Uniform Building Code. Class 1 vessels, equipment and piping will be designed in accordance with the criteria and stresses set forth in the ASME Boiler and Pressure Vessel Code and the ASA B. 31.1 Code for Pressure Piping.

ADEQUACY OF THE SEISMIC DESIGN CRITERIA

We have reviewed the Safety Analysis Report, Amendments, and Addenda. In addition, we have discussed the various aspects of the seismic design of the plant with members of the staffs of the Divisions of Material Licensing and Reactor Licensing at several meetings and with the applicant and members of the staffs at meetings on July 28 and December 18, 1969 and during a site visit on May 22, 1969. We have the following comments regarding the adequacy of the seismic design criteria:

1. The surficial soils at the site consist of loose to medium-dense, fine sands which are underlain at shallow depth by medium-dense to dense clayey sands and sandy clays. The heavier plant structures will be founded at various depths below grade on conventional spread or mat footings. In order to increase the factor of safety against liquefaction of founding soils under earthquake loading, a pad of compacted fill extending 15 feet above ground surface will be placed in the plant area. Settlement in the founding soils under the superimposed load of the compacted fill and structures has been estimated, and the applicant has indicated that the construction schedule will be set so that the underlying soils will pre-consolidate to a considerable degree prior to start of building construction. Allowable soil bearing pressures are presented which should provide for adequate factors of safety against failure.

2. Distinctive topographic features common to this part of the country and to the Savannah River Plant area are the so-called "sinks".

These consist of rounded depressions in the otherwise level ground surface which average 5 feet in depth near the plant site, and range from a few hundred to a thousand feet or more in diameter. Extensive investigations by the U. S. Corps of Engineers have indicated that these sinks are due to subsidence of a zone of soluble calcareous material. Therefore, there is a possibility of settlement of structures resulting from consolidation of soft spots and from collapse of cavities associated with the zone of calcareous materials. Although the possibility of subsidence is small, it could endanger critical structures. Because of the possibility of subsidence at the site, the applicant was requested to exercise extra care in subsurface exploration with particular emphasis on exploration of the possible calcareous zone. After discussion at the December 18, 1969 meeting, the applicant indicated that additional drilling would be carried out at the site. This drilling has been satisfactorily completed and no evidence of voids or calcareous material was found.

3. The metamorphased and folded condition of the Paleozoic and Cambrian basement rocks indicates that the region was once tectonically active. However, the Tertiary and Cretaceous formations are undisturbed and even partly unconsolidated, indicating that deformation has been only slight or non-existent since the Cretaceous. The postulated Triassic trough suggests faulting with vertical displacements prior to the Triassic period. Other similar Triassic troughs on the eastern seaboard were formed in this manner. According to the data submitted by the applicant, there is no evidence to indicate that the Tertiary formations have been displaced by faults in this region. Therefore the trough-forming faults, if they exist, have not been active since at least the Miocene epoch.

4. The Barnwell site is located in the Coastal Plain province of South Carolina, which is characterized by low seismicity with the exception of the Charleston earthquake of 1886, which had an epicentral Modified Mercalli (MM) intensity of X. Areal seismicity, within 150 miles

of the site, is characterized by an apparent random distribution of shocks with varying intensities. The region is geologically stable, and there are no identifiable faults or other recent geologic structures to which earthquake epicenters can be related.

The grestest intensity historically experienced at the site is due to the Charleston earthquake, and is estimated at MM intensity VI to VII+. The applicant has proposed ground surface accelerations of 0.12g and 0.20g for the Operating Basis and Design Basis Earthquakes respectively. We concur with the selection of these accelerations.

5. The applicant has stated that he will use the response spectrum method of dynamic analysis for Class I structures, piping, and equipment, and that discrete-mass multi-degree of freedom mathematical models will be developed for the structural systems. Mathematical models of Class I structures, piping and process systems, and ventilation system and stack will be submitted to the AEC for approval prior to making the seismic analyses. Time-history analyses of the structures will be performed to develop response spectra at the points of support of piping and equipment. Time-histories to be used as input to these analyses will be selected such that they will produce response spectra at least as high as the spectra postulated for the site. We concur in general with this approach. The analytical techniques proposed by the applicant are satisfactory and if properly implemented will result in a conservative design.

6. Since moduli determined from results of seismic surveys for depths from 6 to 45 feet are applicable for low strain levels, we recommend that the applicant demonstrate that these values are applicable for the higher strain levels which could be expected due to earthquake motions (Ref. question 3.2.5). In addition the applicant should justify the method used to establish the moduli for Class I structures on berm fill or on improved soil above elevation - 6.

   This information and justification should be submitted for review and approval prior to utilization of the moduli in the seismic analyses.

7. The applicant has not justified the use of the proposed response spectra as being conservative. The smooth spectra presented for 0.12g earthquake indicates a maximum magnification of about 2.2 times the peak ground acceleration for a damping value of 2%. However, data submitted by the applicant indicate that the spectra should have greater magnification in the period ranges above about 0.2 seconds. In addition, response spectra determined for historic earthquakes indicate a maximum amplification of approximately 2.5 to 4.5 for 2% damping. We do not concur with the applicant's premise that the selected ground accelerations are "very conservative" and that the spectra presented are therefore adequate, or the implication that unconservative response spectra are counterbalanced by overconservative peak acceleration values. It is our opinion that the peak ground accelerations postulated for the OBE and DBE for the site are reasonable but the proposed response spectra are unconservative.

CONCLUSIONS

On the basis of the information presented by the applicant in the Safety Analysis Report, Amendments, and Addenda, and provided that Comments 6 and 7 above are satisfactorily resolved, it is our opinion that the seismic design criteria and approach to seismic design as outlined in the SAR, Amendments and Addenda, if properly implemented by the applicant, will result in a design that is adequate to resist the earthquake conditions postulated for the site.

Roland L. Sharpe

Roland L. Sharpe

Garrison Kost

Garrison Kost

REFERENCES

BARNWELL NUCLEAR FUEL PLANT

(AEC Docket No. 50-332)

Safety Analysis Report, Volumes I and II

Safety Analysis Report, Proprietary Supplement

Amendment dated August 15, 1969

Amendment No. 3

Amendment No. 4

Addenda No. 1, 2, and 3

Law Engineering Report (February 23, 1970)

APPENDIX E

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D.C. 20240

Mr. Harold L. Price
Director of Regulations
U.S. Atomic Energy Commission
Washington, D.C. 20545

JUN 27 1969

Dear Mr. Price:

This will transmit our comments on the application by Allied Chemical Nuclear Products, Inc., for a construction permit for the proposed Barnwell Nuclear Plant, Lower Three Runs Creek, Barnwell County, South Carolina, AEC Docket No. 50-332. These comments are provided in response to Mr. McBride's letter of December 4, 1968.

The plant would be located contiguous to the eastern boundary of the Atomic Energy Commission's existing Savannah River Plant and would process irradiated nuclear power reactor fuel elements consisting of uranium oxide or a mixture of uranium oxide and plutonium oxide. Recovered products would be uranium, plutonium, and neptunium, all as nitrate solutions. The nominal plant capacity would be five metric tons of uranium per day.

Cooling water for the high level liquid waste storage system would be obtained from on-site wells and passed through a closed loop condenser cooling system which includes a cooling tower.

The routine release of radioactive material to the environs via the normal effluents from the plant would be carefully controlled to maintain release rates at the lowest practical value. Gaseous wastes would be vented to the gaseous waste disposal system. Liquid radioactive wastes would be processed, sampled, and analyzed before being released in a controlled manner to Lower Three Runs Creek.

Lower Three Runs Creek, a tributary to the Savannah River, supports largemouth bass, sunfish, and catfish. Historically, this stream supported excellent spawning runs of striped bass. The Savannah River contains largemouth bass, chain pickerel, catfish, crappie, sunfish, as well as spring migrations of striped bass, American shad, and herring.

Lower Three Runs Creek and the Savannah River are flanked by bottomland hardwoods with wooded swamps located near the mouth of the Savannah River. Many species of wildlife including turkey, deer, squirrel, rabbit, bobwhite, wood duck, mallard duck, and black duck inhabit the areas. Fishing and hunting are presently prohibited on

2124

the portion of Lower Three Runs Creek within the boundary of the Savannah River Plant.

The applicant states that the release of radioactive wastes will not exceed maximum permissible concentrations prescribed in Title 10, Part 20, of the Code of Federal Regulations. Although these provisions may safeguard man from undue radiation exposure, they may not always guarantee that fish and wildlife will be protected from adverse effects. If the concentration in the receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge for fish and wildlife. However, radioisotopes of many elements are concentrated and stored by organisms that required these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of food chain just as are the non-radioactive elements. These transfers may result in further concentration of radionuclides and a wide dispersion from the project area, particularly by migratory fish, mammals, and birds.

In view of the extensive fish and wildlife resources in the project area, it is imperative that every possible effort be made to safeguard these resources from radioactive contamination. Therefore, it is recommended that Allied Chemical Nuclear Products, Inc. be required to:

1. Cooperate with the Fish and Wildlife Service, the Federal Water Pollution Control Administration, the South Carolina Wildlife Resources Department, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys including but not limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the effluent outfall.

   b. Beta and gamma radioactivity analysis of selected fish and wildlife species collected as near the effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and submit five copies to the Secretary of the Interior for evaluation prior to project operation.

4. Conduct post-operational radiological surveys similar to those specified in recommendation (2) above, analyze the data,

prepare reports every six months during plant operation or until it has been conclusively demonstrated that no significant adverse conditions exist, and submit five copies of these reports to the Secretary of the Interior for evaluation.

5. Make modifications in project structures and operations to reduce the discharge of radioactive wastes to an acceptable level if it is determined in the pre-operational or the post-operational surveys that the schedule for releases of radioactive effluents would result in harmful concentrations of radioactivity in fish and wildlife.

The opportunity for presenting our views on this subject is appreciated.

Sincerely yours,

Acting Commissioner

APPENDIX F

EVALUATION OF THE FINANCIAL QUALIFICATIONS

We have reviewed the financial information in the application and Amendments No. 1 through 5 thereto of Allied-Gulf Nuclear Services, Allied Chemical Nuclear Products, Inc., and Gulf Energy and Environmental Systems, Inc., for a permit to construct a nuclear fuel reprocessing plant to be known as the Barnwell Nuclear Fuel Plant and to be located in Barnwell County, South Carolina. Based on this information, we have concluded that the applicants are financially qualified to carry out their commitments under the arrangements as stated in the application, as amended, to design and construct the proposed facility.

This conclusion is based upon the following facts and considerations.

1. The applicants, in Amendment No. 5, estimate the costs to construct the Barnwell Plant, including the initial facilities for storage of liquid wastes, to be, at a maximum, $61 million. These costs include contingencies and escalation. The details of these estimates, considered by the applicants as "Company Confidential - Exempt Financial Information," have been reviewed by the Division of Construction and found to be reasonable.

2. Allied Chemical Nuclear Products, Inc., a Delaware Corporation and a wholly-owned subsidiary of Allied Chemical Corporation, and Gulf Energy and Environmental Systems, Inc., a Delaware Corporation and a wholly-owned subsidiary of Gulf Oil Corporation, have formed a

partnership, entitled Allied-Gulf Nuclear Services for the design, construction and operation of a light-water nuclear fuel reprocessing plant (Barnwell) and activities directly related thereto. The arrangements between the parties are spelled out in the Partnership Agreement dated February 1, 1970 (Attachment to Amendment 5). Portions of the agreement (primarily the financial arrangements) contain information considered proprietary. Essentially, these arrangements provide for an equal interest in, and an equal sharing of, the costs and expenses for the design and construction and operation of the Barnwell plant.

The parent company of each partner has guaranteed the performance and assumed each and every liability assumed by its respective wholly-owned subsidiary under the Partnership Agreement. Accordingly, any funds required to finance the design and construction of the Barnwell plant, in the final analysis, will be provided by the respective parent, Allied Chemical Corporation (Allied) or Gulf Oil Corporation (Gulf). As of the present, the respective parents of the partners have authorized the expenditure of funds sufficient to meet the estimated costs of the project through 1970. Allied and Gulf will, at their respective discretions, authorize such further financing as the need for funds for the design and construction of the Barnwell plant arises. (Amendment 5 - page 4)

3. Allied is soundly financed and has plentiful resources at its command. Its sales of products and services totaled over $1.3 billion for 1969. Its Dun and Bradstreet credit rating is AaA1 and Moody's Investors Service rates its Debentures as high-medium grade (A). As of December 31, 1969, the assets of the Corporation totaled $1,524 million; net worth was $750 million; working capital was $289 million; net income for the year was $68 million; and earnings reinvested in operations totaled $416 million. A summary analysis reflecting pertinent financial ratios and a sound financial position as of December 31, 1969 is attached. This analysis is based on the certified financial statements contained in the Corporation's 1969 Annual Report. From our analysis, the staff concludes that there is reasonable assurance that Allied will be able to meet its financial commitments with regard to the design and construction of the Barnwell plant up to the total amount of the estimated costs of the plant, if such should be required.

4. Gulf is one of the largest industrial organizations in the United States. The Corporation is very soundly financed and has plentiful resources at its command. Its sales of products and services during 1969 exceeded $6.1 billion; its Dun and Bradstreet Credit Rating is AaA1; and Moody's rates its Debentures as "gilt-edge" (Aaa). As of December 31, 1969, the assets of the Corporation totaled $8.1 billion; net worth was over

$5.0 billion; working capital was $1.1 billion; net income for the year was $611 million; and earnings reinvested in operations exceeded $3.5 billion. A summary analysis reflecting pertinent financial ratios and a sound financial position as of December 31, 1969 is attached. This analysis is based on the certified financial statements contained in the Corporation's 1969 Annual Report. From our analysis, the staff concludes that there is reasonable assurance that Gulf will be able to meet its financial commitments with regard to the design and construction of the Barnwell plant up to the total amount of the estimated costs of the plant, if such should be required.

ALLIED CHEMICAL CORPORATION
GULF OIL CORPORATION
FINANCIAL ANALYSIS

Docket No. 50-332

(dollars in millions)
Calendar Year Ended December 31

	Allied Chemical		Gulf Oil	
	1969	1968	1969	1968
Working capital	$ 289.2	$ 302.0	$1,088.7	$1,184.4
Current assets	457.3	496.8	2,325.3	2,279.9
Current liabilities	168.1	194.8	1,236.6	1,095.6
Current ratio	2.72	2.55	1.88	2.08
Cash, securities, receivables	261.6	311.9	1,690.9	1,642.3
Acid test ratio	1.56	1.60	1.37	1.50
Long-term debt	387.0	396.9	1,447.5	1,305.3
Net fixed assets	903.4	847.9	5,069.2	4,621.8
Ratio - debt to fixed plant	.43	.47	.29	.28
Net worth	749.7	724.4	5,039.9	4,750.8
Total assets	1,524.0	1,494.6	8,104.8	7,498.3
Proprietary ratio	.49	.48	.62	.63
Total debt (current and fixed)	774.4	770.2	3,064.9	2,747.5
Ratio - worth to debt	.97	.94	1.64	1.73
Net income before interest	95.0	69.4	697.8	675.7
Interest	27.0	28.6	87.2	49.4
No. of times earned	3.52	2.43	8.0	13.7
Net income	68.0	40.8	610.6	626.3
Net worth	749.7	724.4	5,039.9	4,750.8
Rate of return on net worth	9.1%	5.6%	12.1%	13.2%
Total operating expenses	1,224.3	1,199.1	3,749.3	3,423.5
Sales	1,316.1	1,263.1	6,109.9	5,595.7
Operating ratio	.93	.95	.61	.61
Retained earnings	416.0	381.5	3,556.2	3,257.2
Earnings per share of common	$2.44	$1.46	$2.94	$3.02
Total net tangible assets	1,447.1	1,424.9	8,000.4	7,439.0
Long-term debt	387.0	396.9	1,447.5	1,305.3
Ratio	3.74	3.59	5.53	5.70
Moody's Bond Ratings - Debentures	A		Aaa	
Dun and Bradstreet Credit Rating	AaA1		AaA1	

APPENDIX G

SUMMARY TABLE DESCRIBING

THE BASIC OPERATIONS IN THE

BARNWELL NUCLEAR FUEL PLANT

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Cask receiving and handling	Receipt and preparation of shipping cask for unloading	Cask and carrier will be monitored for outside contamination and washed to remove outside dirt. The cask will be removed and the condition of fuel and coolant determined by temperature, pressure, and coolant radioactivity measurements. The cask will be vented to the vessel off-gas system and the primary coolant replaced, if necessary. The cask will be placed in the cask unloading pool where the lid will be removed and fuel elements unloaded remotely under water shield. Empty casks will be decontaminated, monitored, and returned to customer.
Fuel storage and transfer	Storage of fuel elements until dissolution	Fuel element identity will be confirmed and the elements placed in storage canisters in the storage pool. Pool water will be circulated through heat exchangers, inorganic ion exchange beds, and filters to remove fuel decay heat and radioactive contaminants. Elements will be remotely transferred from the pool to the feed mechanism of shear.
Shear	Preparation of fuel for dissolution	Fuel elements will be mechanically chopped into small segments, exposing oxide fuel inside the elements to dissolution while outside cladding (stainless steel or zircaloy) will remain undissolved.

Response 7.1
Page 2

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Dissolution and feed preparation Dissolution	Conversion of the fuel to a liquid solution $3UO_2 + 8HNO_3 \xrightarrow{<8M} 3UO_2(NO_3)_2 + 4H_2O + 2NO$ $UO_2 + 4HNO_3 \xrightarrow{>8M} UO_2(NO_3)_2 + 2H_2O + 2NO_2$ FISSION PRODUCTS + x $HNO_3 \rightarrow$ F.P. $(NO_3)_x$ + yH_2O+z NO $PuO_2 + 4HNO_3 \longrightarrow Pu(NO_3)_4 + 2H_2O$ $NO + 2HNO_3 \xrightarrow{>8M} 3NO_2 + H_2O$	The chopped fuel elements will be contacted with hot, concentrated nitric acid which will convert uranium, plutonium, and most of the fission products to soluble nitrate salts. Undissolved cladding (hulls) will remain in dissolver basket. Gases generated during dissolution will be channeled to off-gas treatment system. Nitrate salt solution will be transferred to tanks for sampling measurement and final acid adjustment.
Solids handling and waste	Disposal of undissolved cladding hulls	The cladding hulls will be rinsed and transferred by shielded trailer to a burial ground. Intermittently, or in case of abnormalities during dissolution, batches of hulls will be checked for complete dissolution of plutonium and uranium.
Co-decontamination and partition cycle	Separation of the plutonium and uranium from the bulk of the fission products and partitioning of the plutonium from the uranium	Adjusted aqueous feed solution and tributyl phosphate (TBP) diluted in a normal paraffin hydrocarbon will be mixed counter-currently in a bank of centrifugal contactors. The organic solution, which preferentially extracts the

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
Extraction	$UO_2^{++} + 2NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} UO_2(NO_3)_2 \cdot 2TBP$ $Pu^{+4} + 4NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} Pu(NO_3)_4 \cdot 2TBP$ $PuO_2^{++} + 2NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} PuO_2(NO_3)_2 \cdot 2TBP$	nitrate complexes of tetravalent plutonium and hexavalent uranium, will exit from the centrifugal contactor and pass through a pulsed scrub column where an aqueous nitric acid solution will remove extracted fission products from the organic stream. The organic stream will pass through a partitioning column where plutonium will be reduced to the inextractable trivalent state and stripped into another aqueous nitric acid stream containing hydrazine. The organic stream will pass through another column where the uranium will be stripped into acidified water.
Reduction and Partitioning	$Pu^{+4}_{aq} + 1e^- \longrightarrow Pu^{+3}_{aq}$ $UO_2^{++}{}_{aq} + 2e^- + 4H^+ \longrightarrow U^{+4}_{aq} + 2H_2O$ $U^{+4}_{aq} + 2Pu^{+4}_{aq} + 2H_2O \longrightarrow UO_2^{++} + 2Pu^{+3} + 4H^+$ $N_2H_4 + 2HNO_2 \longrightarrow N_2 + N_2O + 3H_2O$	
Stripping	$UO_2(NO_3)_2 \cdot 2TBP + H_2O + 2H^+ \rightarrow UO_2^{++}{}_{aq} + 2HNO_3 + H_2O + 2TBP$	
Second uranium cycle	Further decontamination of uranium from fission products	Nitric acid will be added to the aqueous strip stream containing the uranium, and the uranyl nitrate complex will again be preferentially extracted by another TBP solution in a pulsed column. Before leaving the column,

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		the organic stream will be scrubbed successively with strong and dilute nitric acid solutions which remove extracted ruthenium and zirconium-niobium, respectively. Uranium will be stripped from the organic stream in another column, using acidified water, and this solution will be subsequently concentrated by evaporation.
Uranium silica gel, product storage	Final decontamination and disposition of uranium	Concentrated uranium solution will be passed through silica gel beds to remove traces of zirconium-niobium. Uranyl nitrate product solution will be analyzed and stored in tanks until shipment.
Second and third plutonium cycle, storage, and shipping	Final decontamination and disposition of plutonium	Plutonium in aqueous stream leaving partitioning column will be reoxidized to the extractable tetravalent state, which will be preferentially extracted into the TBP organic stream in a pulsed column. In the same column, the organic stream will be scrubbed successively with strong and dilute nitric acid solutions, which will remove extracted ruthenium and zirconium-niobium, respectively. The organic stream will pass through a strip column where plutonium will be reduced to inextractable trivalent state, which will transfer to the aqueous stream of dilute nitric acid and hydrazine. The extraction-stripping
Oxidation	$Pu^{+3} + 2NO_2 + H^+ \longrightarrow Pu^{+4} + HNO_3 + NO$	
Extraction	$Pu^{+4} + 4NO_3^- + 2TBP \xrightarrow{n\text{-}C_{12}H_{26}} Pu(NO_3)_4 \cdot 2TBP$	
Reduction and Stripping	$Pu^{+4}_{aq} + 1e^- \longrightarrow Pu^{+3}_{aq}$ $UO_2^{++}{}_{aq} + 2e^- + 4H^+ \longrightarrow U^{+4}_{aq} + 2H_2O$	

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
	$U^{+4}_{aq} + 2Pu^{+4}_{aq} + 2H_2O \longrightarrow UO_2^{++} + 2Pu^{+3} + 4H^+$	sequence will be repeated in the third plutonium cycle for further decontamination. A TBP scrub stream will
Scrubbing	$TBP + UO_2^{++} + 2NO_3^- \xrightarrow{n\text{-}C_{12}H_{26}} UO_2(NO_3)_2 \cdot 2TBP$	remove residual uranium from the plutonium aqueous stream as it leaves the last strip column. Plutonium concentration will be accomplished by maintaining a high ratio of organic to aqueous flow in the strip columns. Final plutonium nitrate solution will be washed with an organic stream of normal paraffin hydrocarbon (diluent for TBP) to remove traces of TBP and phosphate. Product solution will be analyzed and stored in tanks until shipment. Solvent streams leaving plutonium cycles will pass through a strip column to remove residual inextractable species of uranium and plutonium and will be recycled to the co-decontamination cycle.
No. 1 solvent system Carbonate wash	Removal of degradation products from solvent $Na_2CO_3 + 2(C_4H_9)_2HPO_4 \longrightarrow 2(C_4H_9)_2NaPO_4 + H_2CO_3$ $2RCH_2NO_2 \rightleftharpoons 2RCH = NOOH +$ $Na_2CO_3 \longrightarrow 2RCH = NOONa + H_2CO_3$	Organic solvent stream from co-decontamination and partition cycle will be washed successively with dilute aqueous solutions of sodium carbonate, nitric acid, and sodium carbonate (or sodium hydroxide) to remove organic degradation products by extraction or precipitation; precipitated solids will be removed by a filter. Fresh TBP or diluent (normal paraffin hydrocarbon) will be added,

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		as required, to maintain proper TBP concentration or total solvent inventory.
No. 2 solvent system	Removal of degradation products from solvent	Organic solvent stream from second uranium cycle will be treated similarly to No. 1 system, except the second alkaline wash will be omitted.
Liquid waste treating and storage	Disposal of liquid waste streams with minimum residual waste volume for storage.	The highly radioactive waste stream from the co-decontamination cycle will be concentrated by evaporation; acidity of the concentrated bottoms will be reduced to permit long-term storage in stainless steel tanks by reacting with a sugar solution; overheads will be fed to the low-activity evaporator for further decontamination. Most of the remaining nitric acid waste streams containing low levels of fission products, uranium, and plutonium will be concentrated in the low-activity waste evaporator; concentrated bottoms will be recycled to the co-decontamination cycle; overheads will be condensed and fed to the acid recovery system. Miscellaneous waste streams, containing salts, low levels of fission products and no appreciable uranium or plutonium, will be acidified and concentrated in the general-purpose evaporator; bottoms will be stored; overheads will be monitored for radioactivity content and then
Acid reduction	$C_{12}H_{22}O_{11} + 18.2\ HNO_3 \longrightarrow 12CO_2 + 14.9\ NO + 3.3\ NO_2 + 20.1\ H_2O$	

PROCESS STEP	FUNCTION AND PRINCIPAL CHEMICAL REACTIONS	DESCRIPTION
		discharged.
Nitric acid recovery and storage	Recovery of nitric acid and reduction of nitrogen oxides release to the environs.	Overheads from LAW evaporator will contain most of the tritium (as tritiated water) and some undestroyed nitric acid from the process; they will be condensed and fed to the fractionator which concentrates nitric acid. Recovered acid will be stored and used in make-up of various acid streams; overheads containing tritiated water will be monitored for radioactivity (other than tritium) and released to the stack.
Off-gas treating	Removal of radioactive and other pollutants from gaseous effluents	Off-gas from dissolver will pass through a scrubber where radioactive iodine will be removed by contact with dilute aqueous solution of nitric acid and mercurous/mercuric nitrate; it will subsequently pass through an acid absorber where nitrogen oxides will be removed. Dissolver off-gas and vessel off-gas streams will be combined, passed through another mercurous/mercuric nitrate scrubber, an iodine adsorber bed, and a high-efficiency filter before release to the stack.
Iodine Scrub	$^{*}Hg(NO_3)_2 + 4I^- \longrightarrow HgI_4^{=} + 2NO_3^-$ $^{*}Hg(NO_3)_2 + 3I^- \longrightarrow HgI_3^- + 2NO_3^-$ $^{*}Hg/I_2$ mole ratio ≥ 4	

APPENDIX H

CHRONOLOGY OF REVIEW

BARNWELL NUCLEAR FUEL PLANT - DOCKET NO. 50-332

Date	Event
November 6, 1968	Date of application for Construction Permit.
November 29, 1968	Amendment No. 1, consisting of revised pages plus pages that were omitted from the initial submission of the application.
January 23, 1969	Met with the applicant. The applicant advised us that major process and facility changes were being made to BNFP. The applicant was advised that we would proceed with the evaluation of the application upon receipt of the required preliminary design information about principal safety considerations.
February 5, 1969	Met with the applicant. We discussed changes to BNFP and identified safety issues based upon our preliminary review.
February 24, 1969	Sent to the applicant questions and comments pertaining to seismic and tornado design bases.
February 26, 1969	Sent letter to the applicant advising him that the application did not contain sufficient information to enable us to evaluate the BNFP.
April 11, 1969	Amendment No. 2, consisting of major revisions to the process system, which were discussed on 1/23/69 and 2/5/69.
May 14, 1969	Met with the applicant to discuss proposed modifications to their dissolver design, additional information about their Pu reduction and partition process, and geology considerations at the BNFP.
May 22, 1969	Visit to BNFP site by the applicant, consultants, and Regulatory staff.

June 12, 1969	Met with applicant to discuss preliminary, supplemental information relevant to site visit on May 22, 1969, and nomenclature for BNFP instrumentation.
June 27, 1969	Sent request to applicant for addtional information on liquid effluent control, ground water movement, disposal of fuel element hulls, fuel element storage pool, plutonium product storage, high-level waste storage, and structural design bases.
July 28-29, 1969	Met with applicant and consultants and our consultants in San Francisco, to discuss proposed response to our questions of June 27, 1969, pertaining to seismic and tornado design.
September 5, 1969	Received Amendment No. 3 from applicant, which contained: (a) revised lay-out of facilities and equipment, (b) additional descriptive information about safety systems, and (c) seismic and tornado criteria, which we had previously advised the applicant would not meet our acceptance criteria.
October 2, 1969	Visit to the proposed site by the ACRS Subcommittee, the Regulatory staff and the applicant.
November 5, 1969	Met with the applicant to discuss additional information required to support the application.
November 26, 1969	Sent questions to applicant requesting additional information discussed during meeting on November 5, 1969.
December 17, 1969	Met with applicant and consultants in San Francisco to discuss site drill hole tests, and seismic design bases for the BNFP.
February 3, 1970	Met with applicant to discuss preliminary draft response to our letter dated November 26, 1969

February 16, 1970	Met with applicant to discuss proposed response to our letter dated November 26, 1969. We discussed instrument systems, nuclear criticality safety and ventilation.
March 1, 1970	Met with applicant regarding submittal of portion of the response to our letter of November 26, 1969.
March 2, 1970	Received amendment to the application. (Addendum No. 1, dated February 27, 1970).
March 16, 1970	Received Amendment No. 4 notifying us that Allied Chemical Nuclear Products, Inc., and Gulf General Atomic, Inc., have formed a partnership; Allied-Gulf Nuclear Services.
March 19, 1970	Received amendment to the application. (Addendum No. 2, dated March 16, 1970).
April 10, 1970	Met with the applicant to discuss matters dealing with hydrology, A & E design commitment, seismic response and structural design considerations.
April 16, 1970	Received amendment to the application. (Addendum No. 3, dated April 3, 1970).
April 30, 1970	Received amendment to the application (Addendum No. 4, dated April 28, 1970).
May 15, 1970	Met with applicant to discuss high-activity waste storage and preliminary results of corrosion tests.
June 2, 1970	Visit to Allied-Gulf, Florham Park, N. J., to discuss Quality Assurance Program.
June 3, 1970	Meeting with applicant and ACRS Subcommittee.
June 11, 1970	Received amendment to the application (Addendum No. 5, dated June 9, 1970).
June 22, 1970	Received amendment to the application (Addendum No. 6, dated June 19, 1970).

July 1, 1970	Met with applicant to discuss financial qualifications and National Environmental Policy Act.
July 9, 1970	Met with applicant to discuss preparation for ACRS meeting.
July 10, 1970	Met with ACRS to review the BNFP.
July 20, 1970	Received Applicant's Environmental Report.
August 18, 1970	Received Amendment No. 5 to the application, which contained additional information about financial qualifications.