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STUDIES AND RESEARCH
CONCERNING BNFP

SPENT FUEL DISASSEMBLY AND CANNING PROGRAM
AT BARNWELL NUCLEAR FUEL PLANT (BNFP)
PRELIMINARY PROCESS ASSESSMENT STUDIES

Hans G. Tepp

October 1980

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ABSTRACT

Studies being performed to assess nuclear fuel disassembly and encapsulation to enhance spent fuel storage have not revealed any conditions which unfavorably impact the feasibility of the concept. The studies are aimed at evaluating various issues warranting resolution preliminary to licensing a facility for this spent fuel management concept. The areas assessed are potential accidents and their results; maximum temperature level of canned fuel rods; radiation exposure to personnel during operation and fuel storage; criticality risks during storage and as a result of abnormal incidents; physical security requirements and material accountability measures; the impact of safeguards on economics; and a license schedule projection.

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GLOSSARY

AFR	Away-From-Reactor Fuel Storage Facility
AGNS	Allied-General Nuclear Services
BNFP	Barnwell Nuclear Fuel Plant
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
D/E	Dissassembly and Encapsulation
FRSS	Fuel Receiving and Storage Station
FSP	Fuel Storage Pool
FTP	Fuel Transfer Pool
GVOS	Grade Viewing and Operating Station
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RPC	Remote Process Cell
SNM	Special Nuclear Material

1.0 INTRODUCTION

Preliminary results of studies performed to evaluate aspects of the disassembly and encapsulation of spent nuclear fuel which do not permit physical demonstration to verify feasibility have revealed no conditions that impact the feasibility of the concept. The studies are aimed at matters which would be of interest to licensing a facility for this purpose. As such, the studies are exploring those features of normal and accident conditions which could have an impact on the health and safety of the operating personnel as well as the off-site public. A final report will be prepared at the conclusion of these studies; the conclusions of this preliminary report provide timely justification for continued demonstration of the mechanical process under development.

Spent fuel disassembly is a procedure by which light water reactor fuel assemblies are dismantled into their individual fuel rods (Figure 1-1). The dismantling is accomplished remotely with the fuel in a dry hot cell. Dedicated equipment implements the process. The process feed is horizontal. The process consists of:

- (1) Positioning the intact fuel assembly
- (2) Removing the end fitting
- (3) Pulling the fuel rods
- (4) Gravity collection of the pulled rods
- (5) Vibration assisted pushing of the rods into a can half-section.

The individual fuel rods are encapsulated in a canister to facilitate handling and subsequently stored underwater (Figure 1-2). The fuel disassembly/encapsulation concept (D/E) is recognized as a means of increasing the storage capabilities of fuel pools by significantly reducing the fuel volume. As such, fuel D/E could significantly augment the capacity of an AFR facility. This fuel management concept was studied by Allied-General Nuclear Services (AGNS) under contract to the Department of Energy (DOE) during FY 1979 and FY 1980. Reports (1, 2) detailing the fuel D/E process were issued with an FY 1980 report in preparation.

The earlier reports (1, 2) covered initial reviews of such technical aspects as criticality, thermal considerations, shielding, and potential accidents as they might relate to licensing. A preliminary economics study was also made.

The purpose of the current studies was to expand the scope of these licensing matters and to provide a fuller definition of these technical areas. This report details the preliminary results of the expanded treatment.

FUEL DISASSEMBLY PROCESS FLOW

FIGURE 1-1

CAN LOADED WITH FUEL PINS
FROM TWO COMPLETE PWR FUEL ASSEMBLIES

FIGURE 1-2

2.0 SUMMARY OF RESULTS

2.1 Potential Accidents and Consequences

Analyses of several postulated accidents as a result of equipment malfunction or operating error indicated minimal off-site exposures. They did not impact the feasibility of the process. The accidents were assumed to have caused cladding failure of fuel rods to various degrees, releasing the radioactive fission gases. The most severe accidents assumed total failure of all the rods from two PWR assemblies. The greatest impact on the public at the site boundary was the low dose of 1.4×10^{-2} mrem, equivalent to only 0.02% of 10 CFR 20 limits on an annual basis. Lesser releases incurred even lower body burdens.

The consequences of a severe accident resulting from natural phenomenon, such as earthquake or tornado, normally would not require analysis by virtue of a frequency projection of 10^{-7} /year. However, the impact on the public of the accidents analyzed was found to be insignificant.

The postulated accident with the greatest release potential was structural failure of a full canister in a manner which would release the full complement of fuel rods into the pool. Criticality could result for an idealized fuel rod arrangement in the pool water. However, canister failure from a lengthwise split was shown not to be credible.

2.2 Thermal Analyses

One of the advantages of encapsulating fuel rods is a reduction of pool contamination. If the canister containing the rods could be sealed and remain internally dry, the maximum contamination reduction is realized. It was, therefore, essential that the maximum temperature achieved by the rods be defined.

The thermal analyses found that dry canisters, submerged in water at 38°C (100°F) would show the hottest, interior rod to be 204°C (400°F) when holding rods from two PWR assemblies burned to 33,000 MWd/MTU, and decayed three years. For a canister holding the rods from four BWR assemblies with identical burnup and decay, the hottest pin would achieve 182°C (350°F). These temperature levels are not considered detrimental to the pin structure. However, sudden in-flow of water, as through minor failure of the canister, could cause steam pressure buildup that might deform canister or even jeopardize its fuel rod confinement capability.

The decay time necessary for the inner tubes to remain below 93°C (200°F), and thus conservatively preclude steam generation, was found to be 79 months for the PWR case and 74 months for the BWR case.

An alternative for shorter cooled fuel which would preclude steam pressurization problems would be to flood the canister during positioning

for pool storage. Therefore, thermal analysis was pursued to prove the adequate cooling of the tightly packed fuel rods in a flooded canister. On the basis of a 3-millimeter (1/8 inch) gap between the canister and its rack storage slot and inlet cooling water at 49°C (120°F), the maximum temperature achieved by the water within the flooded canister was found to be 60°C (140°F) for the PWR case. The annular water between the canister and the rack slot would undergo a 6°C (11°F) temperature rise under natural circulation only.

It should be noted that the temperature rise in the annulus is dependent only on the heat load and would, therefore, hold for a dry canister also.

These results indicate that storage pool boiling is precluded for the defined fuel.

2.3 Shielding Studies

The compaction of spent fuel by encapsulating the rods from several fuel assemblies into one canister will intensify the source of radiation (i.e., particles/sec cm³). An assessment of the effect the higher concentration of radioactive material could have on the operating personnel was made using the BNFP remote process cell as a model and indicated that adequate shielding was provided by five-foot, normal density concrete shielding walls for the D/E concept.

The results of the shielding analysis indicate that the dose rate from a full canister is 0.14 mR/hour at the "cold" wall surface of the operating area adjacent to the fuel D/E cell. It was also found that the dose rate from a canister containing rods from two PWR assemblies was essentially equal to that coming from a single, intact PWR assembly. The increased internal attenuation of the fuel materials essentially cancels out the effect of concentrating the radioactivity.

Another analysis was made to determine the dose rate in the operating area for a case in which a feed port for supplying empty canisters provided direct access from the operating station to the RPC. The model placed the 24 centimeter (9.5 inch) square opening above the fuel plane and above the operating personnel, thus, preventing direct streaming. However, the need for supplemental shielding was identified when the canister or intact fuel was centered on the port. Also an acceptable empty canister feed process alternative was identified that resulted in adequate attenuation of the radiation.

Another assessment was made to define the impact on personnel in the storage pool area when encapsulated fuel is stored. The analysis compared the expected dose rate from stored intact fuel assemblies with the expected dose rate from canisters containing rods from two PWR assemblies being positioned in place of the intact fuel.

The analysis concluded that the encapsulated fuel would provide an insignificant dose rate of less than two times the dose rate of intact

fuel (i.e., 5×10^{-4} mR/hr) when both are stored under the nominal 4.4 meters (174 inches) of water cover. The internal attenuation of the closer packed fuel rods helps to offset the effect of higher concentration of activity.

2.4 Criticality

An extensive criticality study was made to verify the nuclear safety of the disassembled and encapsulated fuel, and the details are presented in Appendix A. The study expanded on an earlier study⁽¹⁾ by extending the PWR fuel enrichment to 4.1% for future fuel possibilities. Despite the higher enrichment, a fully packed canister with rods from two PWR assemblies, fully flooded, remained substantially subcritical.

Grossly underpacked canisters missing a majority of rods of fresh PWR fuel at a 4.1% enrichment could cause a problem. However, even at an optimum fuel/water ratio, an underpacked PWR canister will have subcriticality assured for fuel that has a maximum of 3.7% enrichment (i.e. the highest enrichment for existing LWR fuels) without burnup credit. Upon experiencing burnup of 8 to 10,000 MWd/MTU, the theoretical initially 4.1% enriched fuel will also be subcritical.

A flooded canister containing 3.2% enriched BWR will be subcritical even under optimum fuel/water ratio conditions resulting from incomplete packing.

Criticality premised on a hypothetical chain accident which splits the canister lengthwise, releases the fuel rods from a full canister into the storage pool, and by chance, incurs an idealized fuel/water ratio was not supported by analysis. Even though the criticality was theoretically supported, the split of the canister could not be supported as discussed in 3.1.4.

2.5 Seismic Analysis

Current seismic evaluations of the canned fuel, the storage racks and the pool structure, as provided by the AFR model based on BNFP, indicate adequacy. These evaluations are being implemented under the original scope of contract DE-AC09-79ET47921. The seismic criterion used to show this adequacy has been performed using NRC Regulatory Guides 1.60 and 1.61.

2.6 Safeguards

The current study used the BNFP site as a model to define which areas of the site would require physical security measures to assure protection for the fuel D/E area from unauthorized intrusion and sabotage. To this end, all areas which relate to actual operation of the fuel disassembly, encapsulation, and storage were identified. Moreover, the protective umbrella was extended to those areas supporting the operation, such as blower stations, filter niches, and emergency utility systems, necessary to assure safety to the operating personnel and off-site public.

A preliminary assessment was made to define the requirements of material accountability incurred by fuel disassembly. It recognized the requirement of assuring that the disassembled fuel rods be encapsulated without loss of identity. The use of the fundamental tool of item identity requires increased effort during the disassembly-encapsulation phase but is significantly aided during the final storage phase. Several potential alternatives were advanced as accountability methods based on weight balances, nondestructive assay techniques, and video recording of the operations. Heretofore, unattainable assay information resulting from the increased access to all fuel rods essentially individually could provide audit of prior handling as well as enhance fuel accountability; for example, fuel fissile content measurements.

2.7 Economics

This portion of the study concentrated on the impact that imposition of safeguards could have on the capital and operating costs projected for fuel D/E in an earlier report.⁽¹⁾ The added safeguard measures necessary for assured physical protection were estimated to require the addition of \$400,000 of capital funds and \$250,000 per year of operating costs. The preliminary material accountability review did not develop specific cost data, but it is expected that its added costs would be substantially less than those for physical security. On this basis, it is anticipated that the economics of the earlier report⁽¹⁾ are not affected to a significant degree.

2.8 Preliminary Licensing Schedule

A brief study was made to look at the manner by which a license to build and operate a facility for the disassembly and encapsulation of nuclear fuel could be applied for. The results of the study were used to project a preliminary schedule for the license application.

Since a fuel D/E facility by assumption will be part of an AFR facility, the projected schedule depends on the state of the licensing process for the AFR. Three models were studied. One scenario depicted a joint application for both the AFR and fuel D/E facilities. The second model depicted that the fuel D/E license was applied for during a technical review phase of the AFR license application. The third scenario assumed that fuel D/E license was applied for as an amendment to an existing license for the AFR.

The results of the study were that both the joint application and the model which had the fuel D/E applied for during the AFR licensing, were projected to consume about the same time, three to four years. It was concluded that the application for the fuel D/E would disturb the licensing process for the AFR sufficiently to cause a delay of approximately one year.

The third scenario, depicting the license application for the fuel D/E as an amendment to an existing AFR license, could result in the shortest time span. Projections were for as little as six months for review, but a lengthier time period could occur, depending on the decision by the NRC to open the application to public hearings.

3.0 DETAILED RESULTS AND DISCUSSION

3.1 Potential Accidents and Consequences

The potential effects of fuel disassembly and encapsulation on public health and safety were considered in light of various accidents which could be envisioned. Radioactive releases from the fuel rod would occur through breaching of its cladding. Such an event could arise through faults within the rod structure due to irradiation, failure of equipment associated with fuel disassembly, encapsulation, or transport, improper fuel handling, or natural phenomenon. To assess the extent of the releases, several potential causes were assessed as follows.

3.1.1 Physical Condition of Spent Fuel

A major contributing factor to fuel failure is the physical condition of the cladding. Certain changes in the mechanical properties of the cladding materials occur, during irradiation in the reactor, which have a bearing on the ability of the fuel rods to withstand mechanical and thermal stresses. In general, the following qualitative changes have been observed:

- The yield strength of the cladding is increased as a result of irradiation
- The ductility of cladding is decreased
- Irradiation changes the type and pattern of various defects in the metal cladding
- The higher the radiation level the metal is exposed to, the greater the physical effect
- Similarly, the length of exposure of the fuel and the temperature it encounters affect the mechanical properties of the cladding.

During disassembly of spent fuel, it is recognized that routine handling, operations, and potential accidents may result in cracking or failure of the fuel cladding. However, the extent of cladding deterioration and its response to D/E continue under review. To date the literature has not addressed with sufficient detail quantitative determinations of the mechanical properties of spent fuel before and after D/E. These studies of evaluating what impact certain operating or accident stresses may have on fuel integrity will continue through 1981 to attempt a definitive assessment.

In the meantime, however, certain degrees of failure can be assumed from which data can be generated evaluating the impact of fuel cladding failure. The following sections postulate certain likely accidents which could occur during fuel disassembly and encapsulation.

3.1.2 Dropping of Fuel Canisters

One postulated accident study assumed the drop of a full canister (holding the rods from two PWR fuel assemblies) onto a concrete surface. This could occur through failure of the processing equipment that supports the canned fuel. The fall of the can was assumed to have caused all of the rods to suffer sufficient stress to incur loss of cladding integrity. On the assumption that the canister is not gas-tight, all of the radioactive gases in the fuel rod plenums could be released.

Such a release would total close to 3000 Ci of radioactivity to the area, consisting of 5.48 Ci from H-3, 1×10^{-2} Ci from C-14, 2910 Ci from Kr-85, and 3.46×10^{-3} Ci from I-129. These values refer to two PWR assemblies, three year-cooled, and generally described in Table 3-1. The impact of such a release on the public at the site boundary, 2.0 kilometers (1.24 miles) from the point of release, is equivalent to an annual dose to the total body of 1.4×10^{-2} mrem, or 0.02% of 10 CFR 20 limits.

The effects of this mishap are considered very conservative, for it is highly unlikely that all of the fuel rods within a canister would have suffered cladding failure. The packing of the rods which permits relative motion would tend to mitigate the impact forces on at least a portion of the rods removed from the point of impact.

An alternate rendition of a free drop of a full canister could depict the accident to occur over or within the fuel storage pool. It can be assumed that the severity of impact of the can with the pool bottom would be lessened due to buoyancy of the canister and to the frictional resistance of the can with the water. Nevertheless, the worst case could not exceed the release data above since one could take credit for the iodine that would partially dissolve in the water, thus reducing the iodine released to the air. This was assessed in the following case.

This scenario for an accident involving fuel canisters assumes the dropping of a fuel storage rack during transport within the storage pool. One likely configuration of a fuel storage rack is one holding nine canisters per rack unit, with each canister holding the fuel rods from two PWR assemblies. This model visualizes that 10% of the contained rods could be damaged by the impact of the loaded rack. On this basis, the damage is equivalent to 1.8 fuel assemblies (9 canisters x 2 assemblies x 10%).

This damage is essentially the same as for one fuel canister and would release radioactivity as follows.

Upon cladding failure, the fuel rod plenum or gap activity is released to the pool. It was assumed that the tritium, carbon-14 and krypton-85 were not retained in the pool water; however, 99% of the iodine-129 was retained. The overall release of gaseous fission products for these assumptions was then 4.9 Ci from H-3, 9×10^{-3} Ci from C-14, 2.6×10^3

Ci from Kr-85, and 3.2×10^{-5} Ci from I-129. The impact of this release on the public at the site boundary is equivalent to a total annual body dose of 1.6×10^{-3} mrem, and is an insignificant portion of 10 CFR 20 limits of 500 mrem. Minor quantities of other fission products could slowly enter the pool water, cesium for example. However, any release of activity to the atmosphere from this source would be trivial.

Any case of a canister dropping into the pool, however, could present the danger of criticality in the event that the canister splits and permits the rods to tumble out and assume a critical array. An earlier study⁽¹⁾ and the current analysis (Appendix A) concluded that a full canister flooded with water would be subcritical due to insufficient moderation. An accident releasing the rod confinement could result in the rods assuming a physical distribution possessing added moderation more favorable to criticality. This case is discussed at greater length in Section 3.4.

3.1.3 Fuel Damage During Operations

Fuel rods may suffer damage during the mechanical manipulations attending the disassembly of the fuel bundle. One credible accident could occur when a row of fuel pins has been pulled and the rod supports fail prior to release of the pin biters. This could result in one end of the rods dropping 50 cm (20 inches) while the other end is still rigidly supported by the biters. Assuming 100% failure of the cladding of all 17 pins of the extracted row from a PWR assembly, the failure could cause a radioactive release of 3.2×10^{-4} Ci from C-14, 1.8×10^{-1} Ci from H-3, 1.1 Ci from I-129, and 94 Ci from Kr-85. The impact of this release on the public at the site boundary is equivalent to a total body annual dose of 4.4×10^{-4} mrem, or a trivial portion of 10 CFR 20 limits of 500 mrem. The referenced PWR fuel is defined in Table 3-1 and has been cooled three years. The releases are gap gases.

Another credible accident could occur during cutting of the end fitting of a PWR assembly. Through an error in alignment of the cutting head, the cutting device could conceivably cut into several fuel rods, releasing their gap activity. Likewise, one or two fuel rods could fail during rod pulling, due to an undetected cladding defect or embrittlement, and also release the radioactivity in the gap. Assuming the failure of two fuel rods, the following release could occur: 3.8×10^{-5} Ci from C-14, 2.1×10^{-2} Ci from H-3, 1.2×10^{-5} Ci from I-129 and 1.1 Ci from Kr-85. The impact of the release to the public at the site boundary, 2.0 kilometers (1.24 miles) from the point of release, is an annual total body dose 5.2×10^{-5} mrem, or a trivial portion of 10 CFR 20 limits of 500 mrem.

3.1.4 Effects of Extreme Natural Phenomena

An assessment of the effects of accidents which could befall the fuel disassembly and encapsulation facility as a result of extreme natural phenomena, such as earthquake or tornado, suggests that the likelihood of occurrence is minimal and the effects negligible. A frequency

analysis for an earthquake or a tornado for the Barnwell, South Carolina area and resulting system failures found that the combined probabilities are projected at 10^{-7} /yr. In accordance with the NRC Standard Review Plan, occurrences of these frequencies normally do not require further analysis.

The location of the fuel disassembly/encapsulation and storage operations are projected to be implemented in a structure (i.e. the BNFP Separations Facility), which is designed to withstand the effects of these extreme natural phenomena. The projected D/E operating area is the RPC of the facility, having five-foot thick walls of reinforced concrete, adequate to withstand the effects of tornado-generated missiles. Fuel storage is in a recessed pool to minimize the effect of missiles. The portions of the facility related to the fuel D/E and storage are designed to withstand the effects of an earthquake capable of imposing a 0.2 g force. See Sections 3.5 for discussion of the seismic adequacy of the storage pool structure and racks. Thus the structure of the facility supplements the conclusion of the insignificant impact of extreme natural phenomena.

However, although normally unnecessary by current regulatory reviews, the following analyses were prepared. In the event that the minimal chance of an earthquake should occur during movement of fuel or actual disassembly, the fuel per se or the fuel handling equipment could be subjected to undue stress. One can postulate that the fuel could fail at such time. The release of radioactivity for such an accident could be projected to double the maximum quantities defined in Sections 3.1.2 and 3.1.3.

This occurs since a maximum of two (normally one) additional fuel assemblies could be projected in the feed system to the process that could fail all cladding. Also the quake could cause a fuel canister to drop in the pool. If the can then split and the fuel rods fell free in the pool, the assessment of criticality (Section 3.4) concludes that it could assume an idealized fuel rod arrangement. However, analysis of the can indicates that the maximum deceleration force generated with can impact on an unyielding force (i.e., conservative) cannot exceed one third of the force required to split the can lengthwise precluding this occurrence.

3.2 Thermal Analyses

The disassembly of spent fuel and the encapsulation of the fuel rods in canisters having a "double-density" fuel packing raises the question of resulting thermal effects. Increases of temperature for the individual rods, as well as for the filled canister, can be anticipated when this concept is compared to the classic manner of storing intact fuel assemblies. The latter storage manner, with water surrounding the individual rods in a relatively open array, provides an obviously more effective means of removing the heat of radioactive decay. Moreover, the heat generated by the fuel rods impacts on the choice of the manner

by which full canisters can be stored within a pool, i.e., internally "dry" or open to the pool water.

While pool storage of a dry canister is beneficial in preventing contamination of the pool water, it does create a risk should the canister incur a sizeable leak so as to permit entry of substantial quantities of water. In such a case, serious pressurization of the canister could occur due to steam generation as the pool water encounters the hot pins. Moreover, such temperature shocks could affect the structure of the fuel pins.

The thermal analysis was thus pointed at determining the following data:

- (1) Maximum temperature of a fuel rod within a "dry" fuel canister
- (2) The decay time necessary to prevent the hottest rod within a full canister from attaining the boiling point of water.
- (3) The circulating water temperature rise for the canister stored within the fuel rack.

An earlier report on fuel disassembly ⁽¹⁾ briefly addressed this subject of thermal effects and concluded that no harmful effects would be evident with fuel aged 5 years when disassembled rods were encapsulated. The bases for the current thermal analyses were PWR and BWR fuel aged 3 years, as defined in Table 3-1, except that the number of fuel rods for the PWR is 208 per assembly to achieve the densest fuel packing per canister. ⁽¹⁾ The data used in the following assessments were obtained from Reference 3 from which Figures 3-1, 3-2, and Figure 3-3, and Table 3-2 were derived.

3.2.1 Maximum Fuel Pin Temperature - Dry Canister

A canister filled with fuel rods from two PWR assemblies, represented by Figure 1-2, and having an outer wall temperature of 38°C (100°F) is expected to have its hottest fuel pins near the center of the square array. However, assuming a diagonal divider plate of 3 millimeters (1/8 inch), the hottest rods of three-year cooled fuel are displaced by a few rod diameters away from the divider plate due to the latter's heat conduction path. The values for these hottest pins were found to be 204°C (400°F) (See Figure 3-2), and thus would not attain temperature levels considered harmful to pin integrity.

A parallel analysis, in which the canister contains the fuel pins from four-three year cooled BWR assemblies, resulted in the hottest pins attaining a temperature of 182°C (350°F) (Figure 3-3).

These assessments assumed that the filled canisters were dry (no water internally), such as would occur during transport or temporary storage in air at 38°C (100°F) or longer storage in a pool at 38°C (100°F).

3.2.2 Maximum Fuel Pin Temperature - Limited by Decay

The decay time necessary to assure that the hottest fuel rod within a dry canister cannot achieve a temperature equivalent to the boiling point of water can be extracted from Figure 3-2 for PWR and Figure 3-3 for BWR, basis a 38°C (100°F) pool temperature. These figures indicate that the PWR canister is limited to a heat flux of 0.65 kilowatt/fuel assembly, or 1.35 kilowatt/MTU, while the BWR canister is limited to 0.30 kilowatt/fuel assembly, or 1.58 kilowatt/MTU. Projecting these data to Figure 3-1 shows that the PWR fuel would require a decay period of 79 months, and the BWR 74 months, for the hottest rod in their respective canisters to be limited to 93°C (200°F). The limit of 93°C was arbitrarily chosen for conservatism. No credit was taken for the elevated boiling point, 115°C (240°F), of the pool water due to hydraulic head of 8 meters (25 feet) in effect when the fuel is stored within racks at the bottom of the pool.

3.2.3 Circulation Water Temperature Rise - Stored Canister

An alternate manner of storing fuel canisters of disassembled fuel rods permits the canister to fill with pool water through canister design. While this method would cool the rods more effectively by water conduction, it does not minimize pool contamination as would a sealed, dry can. Moreover, compact storage of the canister within racks, requires minimal clearance between the canister and the rack slot to maximize pool capacity.

The thermal study analyzed these effects, using the model of fuel pins from two PWR elements per canister, cooled three years, inlet water at 49°C (120°F) a conservative 3 millimeter (1/8 inch) gap between the canisters within the rack. The results of this study (Table 3-2) showed that the maximum water temperature within the canister to be 60°C (140°F), and that the water within the annulus between the canisters would undergo a temperature rise of 6°C (11°F) with natural circulation. This indicates that sufficient cooling would occur to protect the fuel rods from undue heat.

It should be noted that the temperature rise in the annulus is dependent only on the heat load and would, therefore, hold for a dry canister also.

3.3 Shielding Studies

The handling of spent nuclear fuel for disassembly and encapsulation raises the question of radioactive exposure to operating personnel. The spent fuel is highly radioactive, and it is vital to define the extent of exposure to permit a meaningful evaluation of the fuel disassembly and encapsulation (D/E) concept. Such an assessment was made, using the BNFP as a model of a typical facility for implementation of the D/E concept.

The model depicts that the fuel assemblies are moved from the BNFP Fuel Transfer Pool (FTP) into the Remote Process Cell (RPC) where the fuel is disassembled and the fuel rods encapsulated into canisters. The full canisters are then returned to the FTP for eventual movement to the Fuel Storage Pool (FSP). The operations are viewed and controlled from the Grade Viewing and Operating Station (GVOS), shielded from the RPC by a concrete wall with viewing windows. The process centerline averages approximately four feet from the inside face of this wall and is at an elevation paralleling the personnel in the GVOS.

3.3.1 Dose During Operations

The data for this assessment is based on the fuel characteristics of Tables 3-3 and 3-4 and the shielding properties of five feet of ordinary concrete with a density of 2.037 grams/cm³. The analysis used the MORSE Monte Carlo Radiation Transport code and hand calculations. All shielding calculations include credit for the internal attenuation resulting from materials in the fuel pellets.

3.3.1.1 Wall Attenuation

The initial study located a fuel canister (or fuel assembly) a distance of 1.2 meters (four feet) from the concrete shielding wall and examined the dose rate at a point on the opposite side of the wall at the same elevation as the fuel canister and at its mid-plane (See Figure 3-4). The fuel canister was assumed to contain the rods from two PWR assemblies. The results of the study showed a dose rate at a point just outside the concrete wall from one loaded canister to be 0.14 mR/hour for three-year decayed fuel and 0.035 mR/hour for five-year decayed fuel. The dose values were essentially the same for a single PWR intact assembly as for a loaded canister. Although a fuel canister has twice the source of gammas (i.e., gammas/sec) as the intact assembly (due to having twice as much fuel) the internal attenuation is also higher. These two factors were essentially offsetting.

A situation involving multiple fuel units in the RPC may occur as a result of loss of administrative control. The dose for such a case may be estimated by multiplying the dose value of a single unit by the number of units. For example six units, comprising two fuel canisters and four fuel assemblies, would provide a dose rate of 0.84 mR/hour (6 x 0.14 mR/hour) for three-year cooled fuel. Conservatism exists in such a value for it neglects the "shielding" the individual units could provide relative to a dose point of interest. In such a case, the dose rate would be somewhat lower.

Investigating a generic wall of a four-foot thickness instead of the nominal five-foot thick BNFP wall found that the above dose values were raised by a factor of about 25.

3.3.1.2 Dosage Through an Empty Can Feed Port

One phase of the shielding study addressed the extent of exposure to operating personnel from passing empty canisters into the RPC for subsequent filling with fuel rods. The concept considered an opening in the shielding wall of sufficient size 24 centimeters (9.5 inches) square to permit pass-through of the empty canister. The feed port would run straight through and be perpendicular to the cell wall. The port would be located 2.6 meters (8.5 feet) above the GVOS floor and about 1.1 meters (3.5 feet) above the elevation of any fuel assemblies in the RPC. The elevation of the feed port relative to the fuel is such that there is no "direct" radiation streaming from the fuel to the feed port end in the GVOS. However, the dose rate at the port entrance can be expected to be higher than the values defined in Section 3.3.1.1 due to the effect of scattered radiation through the port.

One processing alternative model for the study is depicted in Figure 3-5 and places the fuel assembly or filled canister with its midplane on the same plane as the center of the feed port. Using the MORSE code, the results indicate a dose rate of about 3000 mR/hour for three-year cooled fuel, at a point near the feed port on the GVOS side of the wall. For five-year cooled, fuel the value is half of this. These high dose rates suggest that this processing alternative may not be used unless supplemental shielding is provided. A model of an acceptable processing alternative is depicted in Figure 3-6 which locates a fuel assembly or canister 5.5 meters (18 feet) away from the feed port during canister feed, mid-point to mid-point. The results bear out a definite improvement, suggesting a dose rate from a single fuel unit of 1-10 mR/hour.

3.3.2 Dose During Pool Storage

Another part of the shielding analysis concerned the dose to which operating personnel would be subjected to once the disassembled but canned fuel rods were returned to and stored within the FSP. Since the canned fuel rods offer a more concentrated source of radiation than intact fuel assemblies, an assessment of the effects of this mode of storage was warranted.

The analysis is based on a study for an AFR facility at the BNFP⁽⁵⁾ which defined the radiation dose from fuel stored at an AFR. The gamma source terms for this study are shown in Table 3-5. Although the terms vary somewhat from those of Tables 3-3 and 3-4, used in the previous sections, the final results of the pool exposure study show such trivial dosage that the minor differences in source terms can be considered to have insignificant impact on the end result.

The results of the AFR study ⁽⁵⁾ show that the dose rate at the surface of the pool at normal water level to be 5.0×10^{-4} mR/hour. The fuel assemblies were shielded by 4.4 meters (174 inches) of water, and the pool contains fuel equivalent to 1200 MTU.

In the case of canned disassembled fuel, there will be approximately twice as much fuel per unit area as in the AFR storage mode. On this basis, the gamma source terms, i.e., gammas/cm³ second, would double resulting in a doubled dose rate of to 1×10^{-3} mR/hour. However, the dose rate would not actually double, because of the greater internal attenuation associated with the higher density storage configuration of the disassembled and encapsulated fuel concept.

Therefore, the dose rate associated with fuel D/E would remain very low, less than two times the calculated AFR dosage of 5×10^{-4} mR/hour.

3.4 Criticality

An earlier report (1) on the criticality aspects of the fuel disassembly and encapsulation concept had concluded that a full complement of canned PWR and BWR fuel rods was safe by undermoderation, with the can flooded and under water.

The preliminary studies were expanded for the current report to include situations in which fewer than a full complement of rods per canister were assumed, such as could be caused by improper operation or accidental loss. Moreover, the maximum enrichment for PWR fuel was raised to 4.1% to cover possible future fuel management schemes. The updated studies are detailed in Appendix A; its abstracted conclusions are as follows:

- (1) A flooded canister with a full complement of rods from two PWR fuel assemblies is still subcritical for the 4.1% enrichment.
- (2) The flooded canister with PWR fuel at 4.1% enrichment would be very close to criticality where an optimum fuel/water ratio is attained through omission of approximately half of the fuel rods depending on the specific fuel design. Subcriticality for the canister is achieved for $\leq 3.7\%$ enrichment even at an optimum fuel/water ratio conditions.
- (3) A flooded can containing BWR fuel rods of 3.2% enrichment will be subcritical even under optimum fuel/water ratio conditions.
- (4) A flooded canister with PWR fuel of initial enrichment of 4.1% will be subcritical at optimum fuel/water ratio conditions after burnup of 8-10,000 MWd/MTU and beyond.
- (5) Under postulated accident conditions resulting in complete loss of confinement of a full complement of fuel rods within a pool, criticality could result for an idealized fuel rod arrangement. However, the canister lengthwise split necessary was shown not to be credible as discussed in 3.1.4.

3.5 Seismic

Studies of the structural interaction between the AFR (1) fuel and rack, (2) rack and the pool mounting structure, and (3) mounting structure and the pool, are being evaluated under both static and dynamic (seismic) situations under the original scope of Contract DE-AC09-79ET47921. The rack and mounting structures have been designed to structurally accommodate both normal (assembled) and canned (disassembled) fuel. The modular rack concept that accommodates both of these fuel conditions has been developed by an experienced rack designer, Nuclear Energy Services of Danbury, Connecticut.

Current evaluations for these conditions indicate that the pool is structurally adequate. The seismic criterion used to show this adequacy has been performed using NRC Regulatory Guides 1.60 and 1.61. The response spectrum was developed by Bechtel Corporation using computerized analyses of maximum fuel loading patterns for OBE and DBE conditions. Spectra for both the shallow fuel storage pool and the deeper fuel transfer and cask unloading pools at the BNFP were developed. These response spectra were utilized as input into a finite element computerized model of the racks and mounting to determine local loadings and stresses. The sizing of the structural members of these components was then determined to assure adequacy during normal and accident conditions.

3.6 Safeguards

The previous reports(1, 2) covering the subject of fuel disassembly and encapsulation did not treat the subject of safeguards, i.e., the protection of the facility from physical intrusion and sabotage, and the material accountability of the fuel. However, regulations require that the subject be dealt with for facility licensing. The current project has made a preliminary assessment of the means of safeguarding a fuel disassembly and encapsulation concept, BNFP as a model.

The assessment assumes that the fuel D/E facility is appendage of an AFR fuel storage facility.

3.6.1 Physical Security

The requirement of physical security for the facility is to assure that the site is secure from intrusion for the purposes of material stealth and diversion and is secure from attempts of sabotage. A thorough analysis of security requirements and the means of achievement were performed for AGNS for an AFR at the BNFP.(4) Since the fuel D/E operating area is a contiguous portion to the fuel storage area, the protected area concept for an AFR would extend over the area of the BNFP site relating to fuel disassembly operations and all related support equipment. This could be done by extending the perimeter fencing and its secure access measures. Included areas would be RPC, the operating aisles serving the RPC, and the cell area adjacent to the RPC serving as an equipment maintenance station and waste processing area intended to

package and temporarily store the fuel assembly skeleton after fuel rod extraction.

In addition to protecting the areas in which the actual fuel disassembly and encapsulation process takes place, vital support areas would need protection. This is necessary to assure that the operations can be performed safely without endangering the health and safety of the operating personnel and the off-site public. Included in this category are the contamination confinement equipment including the blower station, which provides the necessary ventilation, the filter niches serving to clean up the discharged ventilation air, and the facility stack. Moreover, the utility area, providing water, air, steam, and electrical power distribution to the site, would be included in the vital area requiring physical protection.

3.6.2 Material Accountability

The purpose of material accountability and control is to provide information on the source, quantity, and location of special nuclear material and to provide assurance that the material has not been diverted. The traditional and current method for spent fuel assemblies is based on item identification and item control.

The current materials accounting requirements for storage of spent fuel assemblies include records with the assembly identification, location, and Special Nuclear Material (SNM) content and periodic physical confirmation of the identity and location of the assemblies.

The requirement with a fuel disassembly program is to provide the needed assurance that all of the fuel rods from an assembly are contained in a specific canister. When this assurance is provided, the accounting problem parallels the spent fuel storage requirements, with a substantial reduction in the number of items to be accounted for.

The object of the materials accountability study was to explore methods of providing an acceptable level of assurance that all of the spent fuel rods from an assembly are accounted for.

3.6.2.1 Current Spent Fuel Assembly Materials Accountability

The main elements of materials accountability used for intact spent fuel assemblies are:

- (1) A record system which documents receipt, inventory (including storage location), and transfer of all spent fuel in storage.
- (2) An item inventory program which provides for periodic physical inventory of spent fuel assemblies based on item identification.
- (3) A records retention system which protects records by duplication and storage. The retention system also maintains records of transfers for a period of five years after the transfer.

- (4) A reporting system which provides Material Status Reports of special nuclear material contained in spent fuel received, possessed, transferred, disposed of, or lost.
- (5) A reporting system which provides nuclear material transaction reports for receipts and transfers of spent fuel.

3.6.2.2 Requirements

The accounting requirements for spent fuel assemblies apply equally to the spent fuel assemblies prior to disassembly and the canisters containing spent fuel rods after disassembly. The accounting method(s) chosen must provide assurance that all of the spent fuel from an assembly is contained in its specified container. Potential considerations are failure to remove a rod or rods from an assembly because of mechanical reasons, rupture of a rod(s) during disassembly, or failure to transfer pulled rods into their respective canister.

Inability to remove a rod from an assembly could be handled by not continuing the disassembly operation until the rod is removed and inserted in the appropriate container. While this may reduce the number of assemblies that can be processed, it simplifies materials accounting. Alternatively, separate handling could be documented.

Rupture of a rod or rods complicates the accounting program. With a ruptured rod or rods, the possibility of SNM being included with "non-SNM" components, such as the assembly end pieces or skeleton, must be considered. Accounting for the SNM becomes more difficult in this case.

The possibility of spent fuel being included either intentionally or accidentally with "non-SNM" components that exists with intact fuel also exists during the disassembly operation. As a result, a means of assurance that spent fuel is not included with the non-SNM bearing components is needed. This verification could be obtained by nondestructive analysis (NDA) of these components. The NDA technique could be a "go", "no go" based on some minimum detection level for the particular NDA system. The envisioned process enhances the application since interior rods are fully exposed during disassembly and operator monitoring is enhanced by remote viewing. The previous applications of NDA could even be expanded with this increased fuel access to determine heretofore unattainable information such as precise fissile interior content and audits of previous handling.

3.6.2.3 Potential Solutions

Several methods of providing assurance that rods are stored in designated containers are available.

One method of verification that fuel rods are stored in designated containers is a weight balance around the disassembly and canning operation. This might be accomplished by weighing of fuel assemblies and

containers into the disassembly area and of canned rods and non-SNM bearing components out of the disassembly area. Some type of nondestructive assay of the non-SNM bearing components would be required to confirm rods were not diverted via this manner.

Nondestructive assay techniques are available which could provide reasonable quantitative and qualitative assurance that the rods are stored in the appropriate container. While these NDA techniques currently do not provide sufficient accuracy for some accountability measurements, these techniques in the future could be sufficiently accurate to detect the absence of one or more rods. For example, one rod omitted from a canister containing two PWR assemblies would result in a 0.2% material deficiency.

The need to provide verification methods for inspection personnel is of particular concern. One verification method which could be considered is video recording of each entire disassembly and canning operation. The inspector would have control of the camera and tape by the use of tamper-safing seals. This could provide visual assurance that all rods from the fuel assemblies were transferred to its canister and that no rods remain in the stripped skeleton of the assembly.

3.7 Economics

An earlier report⁽¹⁾ on fuel disassembly and encapsulation featured an extensive but preliminary cost estimate of a facility designed for the task. The study defined both the capital and operating costs for such a facility adjoining an AFR site. Since that estimate, various studies and demonstrations of measures designed to improve the safeguarding of nuclear fuel have been completed. Moreover, the preliminary nature of the cost estimate of that earlier report⁽¹⁾ did not attempt to define all the specific support and auxiliary service facilities needed to augment the fuel D/E operations. It was, therefore, deemed useful to review the economics of providing the necessary safeguarding of the actual operating areas for fuel D/E and its ancillary support areas.

Section 3.6 defined which areas of a typical fuel reprocessing facility, exemplified by the BNFP, were needed for the safe and effective operation of the fuel disassembly and encapsulation concept. Moreover, it ascribed to these areas the levels of physical security required and estimated the costs of these measures. The study estimated that imposition of advanced physical security over the fuel D/E operating and support area would add \$400,000 of capital costs and \$250,000 per year of operating costs. The added costs of administering the material accountability phase of safeguards is expected to be well below the costs of physical security, for only a nominal increase in personnel is visualized. Moreover, the inventory records management system is already provided for by the AFR.

Comparing the expenditures incurred by addition of safeguard measures with the \$43.5 million capital monies and the \$1.7 million annual operating cost generated by the earlier study⁽¹⁾, one can accept that

the cost of safeguarding the fuel D/E concept is a small part of the overall cost. The safeguard costs fall well into the limits of accuracy of the initial estimate and, therefore, should not perturb the original economics of fuel disassembly and encapsulation.

The capital costs of fuel D/E will be better defined during FY 1981 when the actual expenditures associated with the demonstration equipment will be closely recorded. The final installed costs can then be compared with the FY 1979 estimate and thereby serve as basis for a cost update in the final report.

3.8 Preliminary Licensing Schedule

A facility handling spent nuclear fuel must, by law, obtain a license from the Nuclear Regulatory Commission before it can be built and operated. This demands that a facility designed to disassemble and encapsulate nuclear fuel must undergo such licensing. Experience has been that these licensing procedures are usually lengthy and complex. Therefore, it is advantageous to develop estimates of the time needed to fulfill the necessary licensing requirements so that licensing can be accomplished in a timely manner. As a result, the task of projecting a licensing schedule was undertaken.

Several cases were modeled, each taking into consideration that the fuel disassembly facility was a companion to an AFR. The differences between the cases were the extent of completion of the licensing process for the AFR before a fuel D/E license was applied for.

3.8.1 Case 1 - AFR Licensing Not Submitted

This model assumes that an application for a license for an AFR facility is not submitted until the licensing of a fuel D/E concept is incorporated. It assumes that the AFR license application is essentially complete, though not yet submitted to the NRC, and that the design of the fuel disassembly and encapsulation is all but completed. The total time needed for the joint licensing process is estimated to be 38-42 months from the time the decision was made to include fuel D/E. Figure 3-7 illustrates the tentative schedule.

3.8.2 Case 2 - AFR License Submitted - Not Yet Granted

This model assumes that a license for an AFR has been submitted and is in the technical review phase by the NRC. It is expected that the fuel D/E application would be in the form of an amendment to the AFR license application. As a result, it could extend the technical review phase of the AFR licensing process by an unknown extent, and is expected to be in the range of 10-12 months. On this basis, the total time for receiving an AFR license is estimated to be 32-48 months, as illustrated by Figure 3-7.

3.8.3 Case 3 - AFR License Approved

This model assumes that the license for an AFR has been issued and an application for the fuel D/E facility is applied for as an amendment. The time required to get this amendment approved could vary from six months to three years. It is expected that the fuel D/E amendment could require full review and public hearings, much like the AFR application and the schedule is illustrated in Figure 3-7.

TABLE 3-1

FUEL AND CANISTER SPECIFICATIONS

	<u>BWR</u>	<u>PWR</u>
Enrichment (% U-235)		
initial (maximum)	3.2	4.1
final (nominal)	0.9	0.9
Maximum fuel burnup (MWd/MTU)	33,000	33,000*
Maximum fuel specific power (MWd/MTU)	35	35*
Cooling time, nominal (years)	3	3
Fuel content/assembly (kilograms/uranium)	190	480
Number of fuel rods/assembly	63	264*
Canister dimensions		
Outside dimension (inches)	9.25	9.25*
Wall thickness (inches)	0.125	0.125
Number of assemblies/canister	4	2

*Unless otherwise specified.

TABLE 3-2

MAXIMUM CANISTER WATER TEMPERATURE AND ANNULUS TEMPERATURE RISE

Decay Heat Rate Per Assembly, kilowatt	Maximum Water Temperature in Canister, °F	Temperature Rise in Canister-Rack Annulus, °F
0.5	112	5
1.0	118	7
2.0	131	10
4.0	154	15

Temperatures calculated with a water inlet temperature of 120°F.

Source: Reference (3)

TABLE 3-3

GAMMA SOURCE TERMS

33,000 MWd/MTU, 3.2% Initial Enrichment, 40 MW/MTU Reactor Power Level.

<u>Energy</u> (MEV)	<u>Three Year Delay</u> (γ /Sec/MTU)	<u>Five Year Delay</u> (γ /Sec/MTU)
0.30	1.3E + 15	4.0E + 14
0.63	1.4E + 16	7.6E + 15
1.10	4.7E + 14	2.4E + 14
1.55	1.5E + 14	6.0E + 13
1.99	3.6E + 13	6.7E + 12
2.38	5.2E + 12	1.3E + 12
2.75	4.0E + 11	1.0E + 11
3.25	1.3E + 10	3.2E + 9
3.70	7.0E + 6	6.4E + 6
4.22	4.4E + 6	4.1E + 6
4.70	2.1E + 6	1.9E + 5
5.25	1.3E + 6	1.2E + 6

TABLE 3-4

NEUTRON SOURCE TERMS *

33,000 MWd/MTU, 3.2% Initial Enrichment, 40 MW/MTU Reactor Power Level.

Three Year Delay
(N/Sec/MTU)

3.5E + 8

Five Year Delay
(N/Sec/MTU)

3.2E + 8

*From spontaneous fissions and α , n-reactions.

TABLE 3-5

GAMMA SOURCE TERMS

29,000 MWd/MTU, 3.2% Initial Enrichment, 32 MW/MTU Reactor Power Level, three-year cooled.

γ Energy, Mev	γ Energy Group No. (MORSE Code)	Source Strength	
		γ /sec/MTU*	γ /sec/cm **
0.30	15	1.09E + 15	1.67E + 09
0.63	12	1.09E + 16	1.67E + 10
1.10	10	3.69E + 14	5.66E + 08
1.55	9	1.15E + 14	1.76E + 08
1.99	8	2.99E + 13	4.58E + 07
2.38	7	4.15E + 12	6.36E + 06
2.75	6	3.19E + 11	4.89E + 05
3.25	5	1.01E + 10	1.55E + 04

*Obtained from the ORIGEN Code.

**Based on source dimensions of 48' x 48' x 12' (active fuel height) for the Fuel Storage Pool and on a total inventory of 1200 MTU.

Source: Reference [3]

THERMAL POWER VERSUS DECAY FROM REACTOR SHUTDOWN

FIGURE 3-1

Source: Reference [3]

MAXIMUM FUEL PIN TEMPERATURE --2 PWR, DRY

FIGURE 3-2

Source: Reference [3]

MAXIMUM FUEL PIN TEMPERATURE -- 4 BWR, DRY

FIGURE 3-3

RPC

GVOS

Loaded Fuel Can or
Fuel Assembly
(Elev. 284')

Dose Point
(Elev. 284')

Concrete Cell Wall

80-196

SHIELDING STUDY MODEL - WALL ATTENUATION

FIGURE 3-4

RPC

GVOS

Loaded Fuel Can or
Fuel Assembly
(Elev. 284')

Feed Port
(9.5" x 9.5"-Elev.
288')

Dose Point
(Elev. 288')

80-196

SHIELDING STUDY MODEL - FUEL CAN AT FEED POINT

FIGURE 3-5

RPC

GVOS

Loaded Fuel Can
or Fuel Assembly
(Elev. 284')

SHIELDING STUDY MODEL - FUEL CAN EIGHTEEN FEET
FROM FEED POINT

FIGURE 3-6

PRELIMINARY LICENSE SCHEDULE

FIGURE 3-7

4.0 REFERENCES

1. Anderson, R. T., Spent Fuel Disassembly: Increasing Hot-Cell Storage Capacity at the Barnwell Nuclear Fuel Plant, Allied-General Nuclear Services, AGNS-35900-1.3-47 (October 1979).
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4. Final System Design Using the AFR Spent Fuel Storage Barnwell Facility as a Model, IAEL-156, International Energy Associates Limited, (for AGNS) (August 1980).
5. Safety Analysis Report, Away-From-Reactor Spent Fuel Storage, Barnwell Facility, Allied-General Nuclear Services (in preparation).

APPENDIX A

CRITICALITY SAFETY STUDY OF FUEL
DISASSEMBLY AND ENCAPSULATION

by

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October, 1980

INTRODUCTION

The purpose of this study is to examine the criticality safety aspects of spent fuel disassembly and encapsulation. In the concept, the fuel rods from two PWR assemblies (or four BWR assemblies) are placed in a square can with an inside dimension of 9.0". The can is assumed to be flooded with water, and water completely surrounds the can.* The can is assumed to be isolated from other nuclear fuel material; that is, the can is not loaded into a fuel storage rack, where neutron interaction could occur with neighboring fuel assemblies. The design basis fuel is assumed to contain zero burnup with an enrichment of 4.1% for PWR's and 3.2% for BWR's.**

There are three basic questions of interest:

- (1) Is there a criticality problem with a full complement of fuel rods in the can under the conditions specified (e.g., 9-inch can, flooded and design basis enrichment)?
- (2) With a full complement of rods in the can, the fuel system will not be at the optimum fuel-to-water ratio from the standpoint of criticality. If fewer rods than a full complement are loaded into a can and the can becomes flooded, the reactivity will increase if the rods are assumed to take on a near uniform lattice. The question arises as to whether a criticality problem can occur with any number of fuel rods in the can, assuming that these rods take on the worst-case arrangement in the can.
- (3) Finally, is there a potential criticality under the hypothetical accident condition involving a complete loss of integrity of the container walls that permits the fuel rods in a can to assume the worst-case configuration.*** Under idealized, worst-case conditions this would result in a full complement of rods (e.g., from two PWR assemblies) being spaced at the optimum fuel-to-water ratio.

* In the "dry" condition, it is impossible to achieve criticality with low enriched uranium. Therefore, the analyses are restricted to the flooded condition as could exist with the loaded fuel can placed in a storage pool.

** These values are consistent with current AGNS design criteria for AFR.

*** This condition is assumed to occur, for example, from the drop of a can in the pool resulting in complete loss of confinement of the fuel rods. This case is examined for study purposes. There is no suggestion here that from a structural standpoint such an event is possible.

EVALUATION

The answers to questions (1) and (3) can be obtained from information in the literature. Question (2) is more difficult and requires specific analyses.

Question 1 - Normal Conditions, Full Complement of Fuel Rods in the Can.

Criticality studies presented in Reference (a) examined the critical and safe dimensions of UO_2 rod lattices in water. In this study the critical and safe cylinder diameters were determined as a function of the UO_2 fuel pellet diameter, the rod spacing,* and the fuel enrichment. The "safe" diameter corresponds to a $K_{\text{eff}} = 0.98$. Figure 1 shows some of the pertinent results from Reference (a).** Specifically, Figure 1 depicts the safe cylinder diameter of a fuel rod lattice as a function of the ratio of fuel pellet volume-to-water volume.*** The safe cylinder diameter is given for fuel pellet diameters of 0.3 inches, 0.4 inches, 0.6 inches, and for enrichments of 3.0% and 5.0%.

From this figure it can be observed that the optimum arrangement of the rod lattice (i.e., the smallest cylinder diameter) occurs at a fuel-to-water ratio in the vicinity of 0.3 to 0.4 for the enrichments of interest. As the fuel-to-water ratio increases beyond this optimum range, the rod lattice becomes progressively undermoderated and, therefore, the safe cylinder diameter increases.

From Figure 1 it is apparent that criticality is not possible for a single fuel can under the conditions where the can contains a full fuel complement.**** The 9-inch square can possesses a cross-sectional area equivalent to a cylinder 25.8 centimeters in diameter. Figure 1 shows

* Rod spacing refers to the center-to-center spacing between fuel rods. For a given enrichment and rod diameter, there is an optimum spacing that yields the smallest critical dimension. If the lattice is too tightly packed, it is said to be undermoderated, if too loose, the condition is said to be overmoderated. For a given set of conditions of the rod lattice, the ratio of the fuel pellet volume to the water volume can be computed, which serves as a useful quantity for study purposes.

** The study reported in Reference (a) assumed no cladding on the fuel pellets. Therefore, the resulting safe dimensions are slightly conservative when applied to the case of clad fuel rods.

*** For the remainder of the report this quantity is designated as the "fuel-to-water ratio".

**** That is, two PWR assemblies or four BWR assemblies.

SAFE CYLINDER DIAMETER VERSUS THE VOLUME RATIO
OF FUEL-TO-WATER IN THE ROD LATTICE

FIGURE 1

that a cylinder of 25.8 centimeters in diameter is safe for enrichments up to 5.0% provided that the fuel-to-water ratio is greater than about 0.65.* As shown on Figure 1, the fuel-to-water ratio of a can containing a full fuel complement is well above the 0.65 value.**

Consequently, in answer to the first question, it can be stated that a single can submerged in water containing a full complement of fuel rods is substantially subcritical.

Question 2 - Optimum Fuel-to-Water Ratio in the Can.

With a full complement of fuel rods in a can, the rod lattice is quite undermoderated. The reactivity of the can will increase if the number of rods is reduced and if the fuel rods are assumed to arrange in a uniform lattice. The question of interest is whether it is theoretically possible to achieve a critical configuration in a 9-inch square can for the design basis fuels - assuming a uniform lattice of rods in the can at the optimum fuel-to-water ratio.

To check out the calculational procedure that was employed to examine this question, a comparison was made of the results from the calculational model with the results reported in Reference (a). Specifically, work presented in Reference (a) determined that the safe cylinder diameter (i.e., $K_{eff} = 0.98$) is 23.8 cm for UO_2 pellets at 4.1% enrichment with the rods located in a uniform lattice arrangement at the optimum fuel-to-water ratio. Results from the calculational model*** for this same configuration and condition yielded a calculated $K_{eff} = 0.978 \pm 0.005$, which is in excellent agreement with the results in Reference (a) (i.e., $K_{eff} = 0.98$).

* This is a conservative assessment of the 9-inch square can since a cylinder is more reactive than a square provided that the two figures have the same cross-sectional area.

** Assuming the fuel from two PWR fuel assemblies per can, the fuel-to-water ratio was determined for the following designs: Westinghouse 15 x 15, Westinghouse 17 x 17; B&W 15 x 15, B&W 17 x 17, CE 14 x 14, and CE 16 x 16. The computed ratios fell in the range 1.51 to 2.25. Assuming the fuel from four BWR assemblies per can, the computed fuel pellet/water volume for the GE 7 x 7 and GE 8 x 8 designs were 1.16 and 1.08, respectively.

*** The calculational model for the study discussed in this section utilized the NITAWL/XSDRNPM/KENO-IV computer codes. The neutron cross section set used was the 123 energy group GAM-THERMOS Library. KENO is a Monte Carlo criticality program, and the K_{eff} results are subject to statistical uncertainties. Unless otherwise noted, the stated uncertainty (i.e., the $\pm$ value) corresponds to one standard deviation. Two standard deviations correspond to a 95% confidence level.

This same calculational approach was used to calculate the Keff of a 9-inch square containing 4.1% enriched UO₂ rods at the optimum fuel-to-water ratio.* Results showed a Keff = 0.991 ± 0.014 , which is sufficiently close to a critical condition that the configuration can not be confirmed to be safe.

As a matter of interest, the above calculation was repeated at the lower enrichment of 3.7%. A value of 3.7% represents the upper end of the enrichment range of actual, existing fuel designs in contrast to the "design basis" value of 4.1% for PWR's. This calculation yielded a Keff of 0.975 ± 0.007 . Based on this result, there is strong evidence that for the worst-case conditions of fuel-to-water ratio the 9-inch square will be subcritical for enrichments $\leq 3.7\%$.

Also of interest is the benefit in terms of reduction in reactivity that can be derived from fuel burnup. To examine this effect, fuel with the design basis initial enrichment of 4.1% was burned in a PWR using the NULIF computer program. From this run the inventories of uranium and plutonium were determined at several selected levels of fuel burnup. Then, the impact of these fuel inventory changes on the reactivity of the 9-inch square array was estimated. These results are shown in Figure 2 for the range of burnup from 0-15,000 MWd/tonne. Over this range the Keff is seen to decrease at a rate of about 0.019 ΔK for each 5,000 MWd/tonne. Results discussed earlier showed that at an initial enrichment of 4.1%, the 9-inch square configuration under worst-case conditions was very close to critical (i.e., calculated Keff = 0.991 ± 0.014). Taking advantage of burnup, this same configuration will be subcritical at the relatively low burnup level of around 8,000-10,000 MWd/tonne (i.e., corresponding to a reduction of approximately 0.03-0.04 in Keff relative to the zero burnup case).

Question 3 - Loss of Confinement

The final question deals with the possibility of criticality under the hypothetical, postulated conditions of complete loss of fuel rod confinement as a result of gross structural can failure. In the worst case, a full complement of fuel rods is visualized as forming a uniform lattice at the optimum fuel-to-water ratio. Under these conditions the fuel rods from the two PWR assemblies form a cylindrical shape with a diameter of about 38 cm, and the rods from four BWR assemblies form a cylindrical shape with a diameter of 35 cm.

* For this calculation the fuel pellet diameter was 0.3815 inches and the Zircaloy cladding was 0.026 inches thick. The enrichment was 4.1% and the uranium fuel density in the pellet was 10.90 grams per cubic cm. The fuel-to-water ratio was 0.315. These specific fuel pellet and clad dimensions correspond to the Westinghouse Zion 2 fuel. However, Figure 1 shows that at the optimum fuel-to-water ratio, the safe dimensions are essentially the same for all fuel pellet sizes in the range 0.3 to 0.6 inches in diameter. Therefore, the calculated results are applicable to all fuel pellet designs of interest.

ESTIMATED CHANGE IN k_{eff} AS A
FUNCTION OF BURNUP (MWD/TONNE)

FIGURE 2

Based on the information in Figure 1, cylindrical shapes of these diameters would be clearly supercritical. Consequently, such an accident must be shown to be incredible on the basis of structural considerations.

CONCLUSIONS

Based on the work summarized herein, the following conclusions were reached:

NORMAL CONDITIONS - FUEL ROD COMPLEMENT

A 9-inch square fuel can submerged in water that contains a full complement of fuel rods (i.e., two PWR or four BWR assemblies) will be well subcritical for the design basis PWR and BWR fuel.

Optimum Fuel-to-Water Ratio

- At the design basis enrichment for PWR's of 4.1%, the 9-inch square can under optimum fuel-to-water ratio conditions is predicted to be very close to critical. Therefore, subcriticality cannot be assured under these conditions. Analysis shows that subcriticality is indicated for enrichments of $\leq 3.7\%$.
- A can containing BWR fuel at the design basis enrichment (of 3.2%) will be subcritical even under optimum fuel-to-water ratio conditions.
- Fuel burnup is effective in reducing the reactivity of the fuel can. Analyses indicate that the 9-inch square can will be subcritical under optimum fuel-to-water ratio conditions containing design basis PWR fuel (i.e., initial enrichment of 4.1%) with fuel burnups in the vicinity of 8,000-10,000 MWd/tonne and beyond.
- Although not specifically addressed in this study, the incorporation of a neutron poison material (e.g., boron) into the diagonal plate within the can should be effective in reducing the reactivity of the system.

Loss of Confinement

- Under severe hypothetical accident conditions resulting in complete loss of confinement of a full complement of fuel rods, a criticality could result for an idealized fuel rod arrangement.

REFERENCES

- (a) Clark, H. K., Critical and Safe Masses and Dimensions of Lattices of U and UO₂ Rods in Water, DP-1014 (February 1966).

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