# ATTACHMENT A

ITEM 1

INCREASE REFUELING BORON CONCENTRATION

DEFINITIONS

REACTOR OPERATING CONDITIONS (Continued)

Cold Shutdown Condition (Operating Mode 4)

The reactor coolant $T_{cold}$ is less than 210°F and the reactor coolant is at shutdown boron concentration.

Refueling Shutdown Condition (Operating Mode 5)

The reactor coolant is at refueling boron concentration and $T_{cold}$ is less than 210°F.

Refueling Operation

Any operation involving the shuffling, removal, or replacement of nuclear fuel, CEA's, or startup sources.

The Refueling Boron Concentration

1900

A reactor coolant boron concentration of at least ~~2300~~ ppm, which corresponds to a shutdown margin of not less than 5% with all CEA's withdrawn. |

Shutdown Boron Concentration

The boron concentration required to make the reactor subcritical by the amount defined in paragraph 2.10.

Refueling Outage or Refueling Shutdown

A plant outage or shutdown to perform refueling operations upon reaching the planned fuel depletion for a specific core.

Plant Operating Cycle

The time period from a Refueling Shutdown to the next Refueling Shutdown.

a. One of the operable charging pumps may be removed from service provided two charging pumps are operable within 24 hours.

b. Both boric acid pumps may be out of service for 24 hours.

c. One concentrated boric acid tank may be out of service provided a minimum of 68 inches of 6-1/4 percent to 12 percent by weight boric acid solution at a temperature of at least 20°F above saturation temperature is contained in the operable tank and provided that the tank is restored to operable status within 24 hours.

d. Only one flow path from the concentrated boric acid tanks to the reactor coolant system may be operable provided that either the other flow path from the concentrated boric acid tanks to the reactor coolant system or the flow path from the SIRW tank to the charging pumps is restored to operable status within 24 hours.

e. One channel of heat tracing may be out of service provided it is restored to operable status within 24 hours.

f. One level instrument on each concentrated boric acid tank may be out of service for 24 hours.

Basis

The chemical and volume control system provides control of the reactor coolant system boron inventory.[1] This is normally accomplished by using any one of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the SIRW storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.

(1) The boric acid pumps can deliver the concentrated boric acid tank contents (6-1/4 - 12 weight percent concentration of boric acid) to the charging pumps. The tanks are located above the charging pumps so that the boric acid will flow by gravity without being pumped.

(2) The safety injection pumps can take suction from the SIRW tank which maintains a boric acid concentration greater than the required refueling concentration.

## 2.0 LIMITING CONDITIONS FOR OPERATION
## 2.3 Emergency Core Cooling System

### Applicability

Applies to the operating status of the emergency core cooling system.

### Objective

To assure operability of equipment required to remove decay heat from the core.

### Specifications

(1) Minimum Requirements

The reactor shall not be made critical unless all of the following conditions are met:

a. The SIRW tank contains not less than 283,000 gallons of water with a boron concentration of at least ~~1800~~ 1900 ppm at a temperature not less than 50°F.

b. One means of temperature indication (local) of the SIRW tank is operable.

c. All four safety injection tanks are operable and pressurized to at least 240 psig with a tank liquid of at least 116.2 inches (67%) and a maximum level of 128.1 inches (74%) with refueling boron concentration.

d. One level and one pressure instrument is operable on each safety injection tank.

e. One low-pressure safety injection pump is operable on each bus.

f. One high-pressure safety injection pump is operable on each bus.

g. Both shutdown heat exchangers and three of four component cooling heat exchangers are operable.

h. Piping and valves shall be operable to provide two flow paths from the SIRW tank to the reactor coolant system.

i. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable. HCV-2914, 2934, 2974, and 2954 shall have power removed from the motor operators by locking open the circuit breakers in the power supply lines to the valve motor operators. FCV-326 shall be locked open.

2.0 LIMITING CONDITIONS FOR OPERATION
2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the reactor vessel head, a pressurizer safety valve, or a PORV is removed.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 312°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 271°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable, a single HPSI pump may be made operable and utilized for boric acid injection to the core.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.

The SIRW tank contains ~~a minimum~~ of 283,000 gallons of usable water containing at least ~~1800~~ 1900 ppm boron(1). This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.(2)

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft$^3$ and the maximum 128.1 inch level corresponds to a volume of 895.5 ft$^3$.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

incident could occur during the refueling operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The shutdown cooling pump is used to maintain a uniform boron concentration.

The shutdown margin as indicated will keep the core subcritical even if all CEA's were withdrawn from the core. During refueling operations, the reactor refueling cavity is filled with approximately 250,000 gallons of borated water. The boron concentraiton of this water (at least ~~1800~~ 1900 ppm boron) is sufficient to maintain the reactor subcritical by more than 5%, including allowance for uncertainties, in the cold condition with all rods withdrawn.(2) Periodic checks of refueling water boron concentration ensure the proper shutdown margin. Communication requirements allow the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above engineered safety features, interlocks are utilized during refueling operations to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. In addition, interlocks on the auxiliary building crane will prevent the trolley from being moved over storage racks containing irradiated fuel, except as necessary for the handling of fuel. The restriction of not moving fuel in the reactor for a period of 72 hours after the power has been removed from the core takes advantage of the decay of the short half-life fission products and allows for any failed fuel to purge itself of fission gases, thus reducing the consequences of fuel handling accident.

The ventilation air for both the containment and the spent fuel pool area flows through absolute particulate filters and radiation monitors before discharge at the ventilation discharge duct. In the event the stack discharge should indicate a release in excess of the limits in the technical specifications, the containment ventilation flow paths will be closed automatically and the auxiliary building ventilation flow paths will be closed manually. In addition, the exhaust ventilation ductwork from the spent fuel storage area is equipped with a charcoal filter which will be manually put into operation whenever irradiated fuel is being handled.(1)

References

(1) ~~F~~USAR, Section 9.5
(2) ~~F~~USAR, Section 9.5.1.2

## 2.0 LIMITING CONDITIONS FOR OPERATIONS

### 2.14 Engineered Safety Features System Initiation Instrumentation Settings (Continued)

(3) Containment High Radiation (Air Monitoring) (Continued)

The setpoints for the isolation function will be calculated in accordance with the ODCM.

Each channel is supplied from a separate instrument A.C. bus and each auxiliary relay requires power to operate. On failure of a single A.C. supply, the A and B matrices will assume a one-out-of-two logic.

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a $\pm 22$ psi uncertainty and was the setting used in the safety analysis.(3)

Closure of the MSIVs (and the bypass valves, along with main feedwater isolation and bypass valves) is accomplished by the steam generator isolation signal which is a logical combination of low steam generator pressure or high containment pressure.

As part of the AFW actuation logic, a separate signal is provided to terminate flow to a steam generator upon sensing a low pressure in that steam generator if the other steam generator pressure is greater than the pressure setting. This is done to minimize the temperature reduction in the reactor coolant system in the event of a main steamline break. The setting of 466.7 psia includes a +31.7 psi uncertainty; therefore, a setting of 435 psia was used in the safety analysis.

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of approximately 24 minutes following a safety injection signal. The switchover point of 16 inches above tank bottom is set to prevent the pumps from running dry during the 10 seconds required to stroke the valves and to hold in reserve approximately 28,000 gallons of at least ~~1000~~ 1900 ppm borated water. The FSAR loss of coolant accident analysis(4) assumed the recirculation started when the minimum usable volume of 283,000 gallons had been pumped from the tank.

## 4.0 DESIGN FEATURES

### 4.4 Fuel Storage

#### 4.4.1 New Fuel Storage

The new unirradiated fuel bundles will normally be stored in the dry new fuel storage rack with an effective multiplication factor of less than 0.9. The open grating floor below the rack and the covers above the racks, along with generous provision for drainage, precludes flooding of the new fuel storage rack.

New fuel may also be stored in shipping containers or in the spent fuel pool racks which have a maximum effective multiplication factor of 0.95 with Fort Calhoun Type C fuel and unborated water.

The new fuel storage racks are designed as a Class I structure.

#### 4.4.2 Spent Fuel Storage

Irradiated fuel bundles will be stored prior to off-site shipment in the stainless steel lined spent fuel pool. The spent fuel pool is normally filled with borated water with a concentration of at least ~~1800~~ ppm. 1900

The spent fuel racks are designed as a Class I structure.

Normally the spent fuel pool cooling system will maintain the bulk water temperature of the pool below 120°F. Under other conditions of fuel discharge, the fuel pool water temperature is maintained below 140°F.

The spent fuel racks are designed and will be maintained such that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) assuming the pool is flooded with unborated water. The racks are divided into 2 regions. Region 1 racks are surrounded by Boraflex; Region 2 racks have no poison. Acceptance criteria for fuel storage in Regions 1 and 2 are delineated in Section 2.8 of these Technical Specifications.

ITEM 2

SUSPEND CERTAIN SAMPLING WHEN ALL FUEL HAS
BEEN REMOVED FROM THE REACTOR VESSEL

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.8 Refueling Operations

**Applicability**

Applies to operating limitations during refueling operations.

**Objective**

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

**Specifications**

The following conditions shall be satisfied during any refueling operations:

(1) The equipment hatch and one door in the air lock shall be properly closed. In addition, all automatic containment isolation valves shall be operable or at least one valve in each line shall be closed.

(2) The five containment atmosphere and plant ventilation duct radiation monitors that initiate closure of the containment pressure relief, air sample, and purge system valves shall be tested and verified to be operable immediately prior to refueling operations. The five monitors shall employ one-out-of-five logic from separate contact outputs for VIAS.

(3) Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

(4) Whenever core geometry is being changed, neutron flux shall be continuously monitored by at least two source range neutron monitors, with each monitor providing continuous visual indication in the control room. When core geometry is not being changed, at least one source range neutron monitor shall be in service.

(5) At least one shutdown cooling pump and heat exchanger shall be in operation. However, the pump and heat exchanger may be removed from operation for up to one hour per 8 hour period during the performance of core alterations in the vicinity of the reactor coolant hot leg loops or during manipulation of a source.

Delete (6) During reactor vessel head removal and while refueling operations are being performed in the reactor, the refueling boron concentration shall be maintained in the reactor coolant system and shall be checked by sampling on each shift. Delete

6 ~~(7)~~ Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

7 ~~(8)~~ When irradiated fuel is being handled in the auxiliary building, the exhaust ventilation from the spent fuel pool area will be diverted through the charcoal filter.

8 ~~(9)~~ Prior to initial core loading and prior to refueling operations, a complete check out, including a load test, shall be conducted on fuel handling cranes that will be required during the refueling operation to handle spent fuel assemblies.

9 ~~(10)~~ A minimum of 23 feet of water above the top of the core shall be maintained whenever irradiated fuel is being handled.

10 ~~(11)~~ Storage in Region 1 and Region 2 of the spent fuel racks shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.0 weight percent of U-235.

11 ~~(12)~~ Storage in Region 2 of the spent fuel racks shall be restricted to those assemblies whose parameters fall within the "acceptable" region of Figure 2-10.

If any of the above conditions are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made. However, refueling operations may commence and continue with less than 5 containment atmosphere and plant ventilation duct radiation monitors provided that gross, particulate and iodine monitors are monitoring the stack effluent. These three plant ventilation duct radiation monitors will initiate closure of the containment pressure relief, air sample and purge system valves and shall employ a one-out-of-three logic for the initiation of VIAS.

Irradiated fuel movement shall not be initiated before the reactor core has decayed for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power.

Basis

The equipment and general procedures to be utilized during refueling operations are discussed in the USAR. Detailed instructions, the above specifications, and the design of the fuel handling equipment incorporating built-in interlocks and safety features provide assurance that no

TABLE :-4 (Continued)

MINIMUM FREQUENCIES FOR SAMPLING TEST

| | Type of Measurement and Analysis | Sample and Analysis Frequency |
|---|---|---|
| 1. Reactor Coolant (Continued) | | |
| (c) Cold Shutdown (Operating Mode 4) | (1) Chloride | 1 per 3 days |
| (d) Refueling Shutdown (Operating Mode 5) | (1) Chloride<br>(2) Boron Concentration | 1 per 3 days (3)<br>1 per 3 days (3) |
| (e) Refueling Operation | (1) Chloride<br>(2) Boron Concentration | 1 per 3 days (3)<br>1 per shift (3) |
| 2. SIRW Tank | Boron Concentration | 1 per 31 days |
| 3. Concentrated Boric Acid Tanks | Boron Concentration | 1 per 31 days |
| 4. SI Tanks | Boron Concentration | 1 per 31 days |
| 5. Spent Fuel Pool | Boron Concentration | 1 per 31 days |

(1) Until the radioactivity of the reactor coolant is restored to ≤ 1  Ci/gm DOSE EQUIVALENT I-131.

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was subcritical for 48 hours or longer.

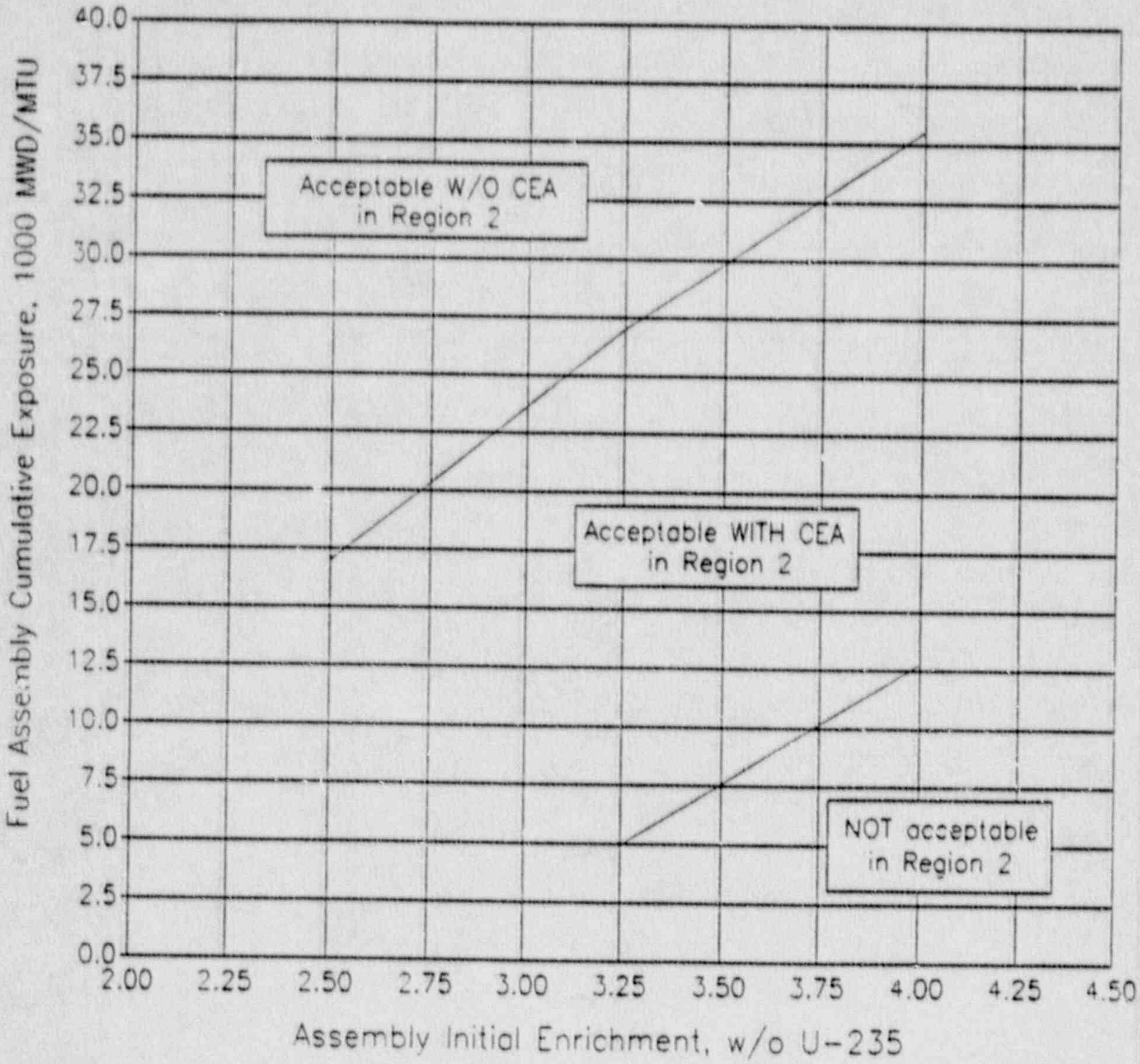
(3) Boron and Chloride sampling/analyses are not required when the core has been off-loaded. Reinitate boron and chloride sampling/analyses one shift prior to reintroduction of fuel into the cavity to assure adequate shutdown margin is maintained.

ITEM 3

REVISE STORAGE REQUIREMENTS FOR
SPENT FUEL POOL REGION 2

# SPENT FUEL POOL REGION 2 STORAGE CRITERIA

Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region 2




| SPENT FUEL POOL REGION 2 STORAGE CRITERIA | OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION-UNIT No.1 | FIGURE 2-10 |
|---|---|---|

Amendment No.

ITEM 4

ALLOW DIRECT TRANSFER OF SPENT FUEL
FROM THE REACTOR CORE TO THE
SPENT FUEL POOL REGION 2

(7) Direct communication between personnel in the control room and at the refueling machine shall be available whenever changes in core geometry are taking place.

(8) When irradiated fuel is being handled in the auxiliary building, the exhaust ventilation from the spent fuel pool area will be diverted through the charcoal filter.

(9) Prior to initial core loading and prior to refueling operations, a complete check out, including a load test, shall be conducted on fuel handling cranes that will be required during the refueling operation to handle spent fuel assemblies.

(10) A minimum of 23 feet of water above the top of the core shall be maintained whenever irradiated fuel is being handled.

(11) Storage in Region 1 and Region 2 of the spent fuel racks shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.0 weight percent of U-235.

(12) Storage in Region 2 of the spent fuel racks shall be restricted to those assemblies whose parameters fall within the "acceptable" region of Figure 2-10.

If any of the above conditions are not met, all refueling operations shall cease immediately, work shall be initiated to satisfy the required conditions, and no operations that may change the reactivity of the core shall be made. However, refueling operations may commence and continue with less than 5 containment atmosphere and plant ventilation duct radiation monitors provided that gross, particulate and iodine monitors are monitoring the stack effluent. These three plant ventilation duct radiation monitors will initiate closure of the containment pressure relief, air sample and purge system valves and shall employ a one-out-of-three logic for the initiation of VIAS.

Insert

Irradiated fuel movement shall not be initiated before the reactor core has decayed for a minimum of 72 hours if the reactor has been operated at power levels in excess of 2% rated power.

Basis

The equipment and general procedures to be utilized during refueling operations are discussed in the USAR. Detailed instructions, the above specifications, and the design of the fuel handling equipment incorporating built-in interlocks and safety features provide assurance that no

INSERT

The spent fuel assembly may be transferred directly from the reactor core to the spent fuel pool Region 2 provided the independent verification of assembly burnups as defined in Special Procedure SP-BURNUP-1 has been completed and the assembly burnup meets the acceptance criteria identified in Technical Specification Figure 2-10.

ITEM 5

DELETE REQUIREMENT FOR FUEL
PERFORMANCE REPORT AT END
OF EACH CYCLE

5.9.3 Special Reports

Special reports shall be submitted to the Regional Administrator of the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.

b. Tendon surveillance, reference 3.5.

c. Containment structural tests, reference 3.5.

d. Special maintenance reports.

e. Containment leak rate tests, reference 3.5.

f. Radioactive effluent releases, reference 2.9.

g. Materials radiation surveillance specimens reports, reference 3.3.

~~h. Fuel performance following each refueling outage.~~

h. ~~i.~~ Fire protection equipment outage, reference 2.19.

i. ~~j.~~ Post-accident monitoring instrumentation, reference 2.21.

5.9.4 Unique Reporting Requirements

a. Radioactive Effluent Release Report

A report covering the operation of the Fort Calhoun Station during the previous six months shall be submitted within 60 days after January 1 and July 1 of each year per the requirements of 10 CFR 50.36a.

The radioactive effluent release report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in Regulatory Guide 1.21, Revision 1.

The radioactive effluent release report shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter as outlined in Regulatory Guide 1.21, Revision 1.

The radioactive effluent release report shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21, Revision 1. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The meteorological conditions concurrent with the

# ATTACHMENT B

ITEM 1

INCREASE REFUELING BORON CONCENTRATION

## DISCUSSION OF CHANGE

The proposed amendment to the Technical Specification increases the reactor vessel refueling boron concentration from 1800 to 1900 ppm. The increased concentration is required to maintain adequate shutdown margin for Cycle 13.

The refueling boron concentration is defined to provide adequate shutdown margin and provide adequate operation response time in the event of an inadvertent dilution event. The shutdown margin and boron concentration requirements are determined using OPPD methodology outlined in the OPPD Topical Report OPPD-NA-8303 Rev. 02 (Reference 1). The boron dilution event analysis ensures a dilution to critical time of not less than 15 to 30 minutes (mode dependent) assuming the maximum credible influx of unborated water.

The requirements of the refueling boron concentration are to provide:

1) a dilution time to critical not less than 30 minutes, and

2) a shutdown margin of not less than 5% assuming a freshly loaded core with all CEAs withdrawn, in accordance with the Technical Specification definition of refueling boron concentration.

For Cycle 13, the current Technical Specification refueling boron concentration of 1800 ppm meets neither of the above criteria. A revised refueling boron concentration of 1900 ppm has been shown to be adequate to fulfill both of the above requirements. The attached table summarizes the results of the boron dilution analyses for Cycles 12 and 13 for the Refueling Mode (Mode 5).

## JUSTIFICATION

Increasing the reactor vessel refueling boron concentration to 1900 ppm will maintain an adequate shutdown margin for Cycle 13.

TABLE 1

REFUELING MODE INPUTS AND RESULTS OF THE BORON DILUTION EVENT

| Parameter | Cycle 12 | Cycle 13 |
|---|---|---|
| Critical Boron Concentrations, All Rods Out, Zero Xenon (ppm) | 1400 | 1454 |
| Inverse Boron Worth Assumed in Dilution to Critical Calculation (ppm/%$\Delta\rho$) | -55 | -55 |
| Time to Loose Prescribed Shutdown Margin (min.) | 31.2 | 33.3 |
| Refueling Boron Concentration | 1800 | 1900 |
| Actual Shutdown Margin (%) | 5.0 | 5.2 |

## NO SIGNIFICANT HAZARDS

The proposed amendment to the Technical Specifications will increase the reactor vessel refueling boron concentration from 1800 to 1900 ppm.

The Technical Specification document changes required are found in Sections 2.2, 2.3, 2.8, 2.14 and 4.4 on pages 2, 2-18, 2-20, 2-22, 2-39, 2-62, and 4.4.

The refueling boron concentration for Cycle 13 is defined to provide adequate shutdown margin and provide adequate operation response time in the event of an inadvertent dilution event. The shutdown margin and boron concentration requirement is determined using OPPD methodology outlined in the OPPD Topical Report OPPD-NA-8303, Rev. 02 (Reference 1). The boron dilution event analysis ensures a dilution to critical time of not less than 15 to 30 minutes (mode dependent) assuming the maximum credible influx of unborated water.

## BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve an increase in the probability or consequences of an accident previously evaluated. The revised refueling boron concentration ensures the existence of both a 5% $\Delta\rho$ or greater shutdown margin with all CEAs withdrawn from the core and a dilution time to critical which is greater or equal to 30 minutes. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the existing Technical Specification controls prevents the possibility of a new or different kind of accident.

3) Involve a reduction in the margin of safety. Specifications involving the minimum refueling shutdown margin are maintained above the minimum required margin and the dilution time to critical conforms to current plant conditions and, therefore, preserves the margin of safety. Increasing the boron concentration ensures that the minimum shutdown margin is maintained and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

ITEM 2

SUSPEND CERTAIN SAMPLING WHEN ALL FUEL HAS BEEN REMOVED FROM THE REACTOR VESSEL

## DISCUSSION OF CHANGE

The proposed amendment to the Technical Specification allows for suspension of boron sampling in the reactor vessel when all fuel has been removed. Sampling will be reinitiated prior to reintroduction of the fuel into the reactor vessel to insure adequate shutdown margin.

Suspension of boron sampling of the reactor vessel coolant when all fuel is removed will not affect the plant safety since no fuel is present.

The reactor vessel coolant boron concentration requirement is based on the need for adequate shutdown margin when fuel is present. When all the fuel is removed, the need for boron is eliminated and hence the need for sampling is eliminated. Elimination of the sampling requirement for the reactor vessel head removal will not adversely impact the safe operation since the shutdown margin calculations do not credit the CEA's. The intent of the reactor vessel head removal would be to ensure CEA's were not inadvertently withdrawn causing a criticality excursion, however since the refueling shutdown calculations include an all rods out assumption, then the deletion of the boron shift sampling requirements will not change the safety analyses. A sampling frequency of once per 3 days would be consistent with refueling shutdown conditions.

The deletion of the chloride sampling will not adversely impact the fuel since the purpose of maintaining the chloride chemistry level is to meet warranty obligations of the fuel vendor and reduce the possibility of intergranular stress corrosion cracking in the fuel assembly material. The chloride chemistry level is established to prevent any potential degradation of the fuel mechanical design properties or RCS piping. When fuel is not present in the reactor cavity a sampling frequency of once per 3 days is consistent with refueling shutdown conditions. The chloride chemistry level of the fuel assemblies is met by sampling of the Spent Fuel Pool.

## JUSTIFICATION

The suspension of reactor vessel coolant boron sampling or chloride sampling when all fuel is removed from the vessel does not compromise or affect the safety of the plant operation.

## NO SIGNIFICANT HAZARDS

The proposed amendment to the Technical Specification allows for suspension of boron and chloride sampling in the reactor vessel when all fuel has been removed.

The Technical Specification document changes required are contained in pages 2-37, 2-38, and 3-19 of Sections 2.8 and 3.2.

The reactor vessel coolant boron concentration requirement is based on the need for adequate shutdown margin when fuel is present. When all the fuel is removed, the need for boron is eliminated and hence the need for sampling is eliminated. Elimination of the sampling requirement for the reactor vessel head removal will not adversely impact the safe operation since the shutdown margin calculations do not credit the CEAs. The intent of the reactor vessel head removal would be to ensure CEAs were not inadvertently withdrawn causing a criticality excursion, however since the refueling shutdown calculations include an all rods out assumption, then the deletion of the boron shift sampling requirements will not change the safety analyses. A sampling frequency of once per 3 days would be consistent with refueling shutdown conditions.

The deletion of the chloride sampling will not adversely impact the fuel since the purpose of maintaining the chloride chemistry level is to meet warranty obligations of the fuel vendor and reduce the possibility of intergranular stress corrosion cracking in the fuel assembly material. The chloride chemistry level is established to prevent any potential degradation of the fuel mechanical design properties or RCS piping. When fuel is not present in the reactor cavity a sampling frequency of once per 3 days is consistent with refueling shutdown conditions. The chloride chemistry level of the fuel assemblies is met by sampling of the Spent Fuel Pool.

## BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change allows for the suspension of boron and chloride sampling during the time the fuel is removed from the reactor vessel with no changes in specifications. Since the fuel source is removed, shutdown margin in the reactor vessel is not required and hence boron sampling is not required and the mechanical design properties of the fuel or RCS piping are not subject to potential degradation due to intergranular stress corrosion cracking potentially induced by a high chloride level. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The use of the proposed revised Technical Specification controls will not result in the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Specifications involving the boron sampling ensure that the shutdown margin conforms to current plant conditions and, therefore, preserves the margin of safety. Since the fuel source is removed, shutdown margin in the reactor vessel is not required and hence boron sampling is not required. The fuel manufacturer's chloride chemistry requirements are met by sampling of the Spent Fuel Pool during the period the core is offloaded. This maintains the mechanical design properties of the fuel. Consolidation of all the boron and chloride sampling requirements in one location in the Technical Specifications ensures compliance of sampling requirements and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

ITEM 3

REVISE STORAGE REQUIREMENTS FOR
SPENT FUEL POOL REGION 2

## DISCUSSION OF CHANGE

The proposed amendment to the Technical Specifications will allow spent fuel assemblies with limited exposure to be stored in Region 2 of the spent fuel pool, if a full length Control Element Assembly (CEA) is inserted in the fuel assembly prior to the move to Region 2.

The spent fuel storage racks at the Fort Calhoun Station are organized into two regions. Either new fuel or discharged fuel may be stored in Region 1, but only spent fuel meeting the minimum exposure requirements of Technical Specifications Figure 2-10 may be stored in Region 2.

The computer code selection, qualification and analysis were performed by Pickard, Lowe and Garrick.

The computer codes used were thoroughly qualified by comparison with measured critical experiments and other published data. These codes and the qualification performed are described below.

### LEOPARD

The LEOPARD code determines fast and thermal spectra and cell averaged cross-sections, using only geometric, material composition and temperature data. The code models the fission product cross-sections for depleted fuel after removal from the reactor, and provides a direct interface to PDQ and CINDER.

### BLACKCYL

The BLACKCYL code uses LEOPARD-generated microscopic cross-sections and geometric data to produce four-group cell-averaged cross-sections for the CEAs for PDQ.

### CINDER

The CINDER code is a zero-dimensional depletion code used to determine the behavior of fission products as a function of time both during the power phase and after shutdown. After shutdown, the decay of fission products causes the average fission product absorption cross-section to vary, falling initially but rising monotonically after approximately half of a year until the end of the time for which calculations were performed (40 years). The stable isotopes accumulated after 40 years precludes the possibility of a decrease below the minimum value even after 40 years. The primary quantities provided by CINDER for spent fuel rack criticality analysis are the average absorption cross-sections for the decaying fission products when these are a minimum, since this maximizes the reactivity of the fuel.

A preprocessor (INCIND) and a post-processor (CORR34) are used with CINDER. INCIND converts a LEOPARD output file into a CINDER input file. CORR34 determines the ratio of the fission product absorption cross-sections generated by CINDER to those produced by LEOPARD at the same fuel exposure.

PDQ

The PDQ code is a diffusion theory code used to determine the $k_\infty$ of the rack using cross-sections generated directly or indirectly from LEOPARD or BLACKCYL. PDQ can be used for resolution of one-dimensional problems.

2.1.2 Qualification

The LEOPARD, PDQ, and BLACKCYL codes have been qualified by the following extensive series of benchmarks against measured criticals:

- Fourteen Westinghouse $UO_2$ Zircalloy-4 clad cylindrical core critical experiments with 2.719 w/o U-235 fuel.[1]
- Five Battelle criticals with 2.35 w/o U-235 fuel.[2]
- Five Saxton criticals with mixed oxide fuel.[3]
- Six Esada criticals with mixed oxide fuel.[4]
- Ten Battelle criticals with 4.31 w/o U-235 fuel.[5]
- Five Battis criticals with cylindrical absorbers.[6]

The first five sets of criticals were calculated using LEOPARD and PDQ only, while the final set required the use of BLACKCYL as well. The mean calculated $k_\infty$ of these 45 measured criticals is 0.9947 and the standard deviation is 0.0041. This results in a bias in the results of 0.0053 and a 95/05 uncertainty of 0.0086.

LEOPARD calculated results were compared to measured values of fuel composition as a function of burnup for the Yankee Core I and II and Saxton Core II spent fuel. LEOPARD accurately traced the actual fuel composition as a function of exposure. CINDER and LEOPARD results were shown to be in close agreement. CINDER calculated values of fission product concentrations were in good agreement with the activities for an average end-of-life core given in the WASH-1400 Reactor Safety Study.

Base Case

The criticality analysis of the spent fuel pool storage rack was performed by:

1. Using LEOPARD to calculate the condition of the depleted fuel with enrichments of 3.25 w/o &-235 at several different exposure levels.
2. Using INCIND, CINDER and CORR34 to compute the minimum fission product absorption cross-sections at selected exposures for each enrichment.

3. Using LEOPARD, with input from the depletion runs and the fission product absorption cross-sections, to compute the cell-averaged cross-sections for the fuel.
4. Using LEOPARD to compute the cross-sections for the water and stainless steel present in the rack.
5. Using LEOPARD and BLACKCYL to compute the cell-averaged cross-sections for the control element assembly cells.
6. Using PDQ along with geometric information and the cross-sections generated above to calculate the $k_\infty$ for the rack, with the CEAs in place, for the enrichments and exposures used above.

This analysis yields the $k_\infty$ of the rack as a function of enrichment and exposure, before biases, uncertainties, and accident conditions are considered.

Calculation Results

The calculations performed for this analysis included the determination of the minimum amount of $B_{10}$ remaining in the CEAs after their exposure in the core and the determination of the minimum allowable fuel exposure as a function of initial enrichment for an assembly with an used CEA in place to be allowed into Region 2 of the spent fuel pool.

The density of $B_{10}$ in the control element assembly is large enough that it essentially depletes from the outside to the inside. This is so because the thermal disadvantage factor for a fresh control element at hot conditions is more than 50, and the absorption mean free path of thermal neutrons in the control element assembly is 0.01 cm. Thus, as the $B_{10}$ is burned out of the control element, it is assumed that the radius of the absorber decreases, while the number density of $B_{10}$ in the remaining central portion remains constant.

A series of LEOPARD, BLACKCYL, and PDQ calculations were performed to determine the rate at which neutrons were absorbed in the control element as a function of the amount of $B_{10}$ remaining. Separate calculations were performed for the four outer rods of the CEA and the single center rod. Since virtually all of the absorption is by the $B_{10}$, this gives a differential equation for the amount of $B_{10}$ as a function of exposure. This differential equation may be solved approximately by assuming that the rate of absorption varies in a piecewise linear fashion between the data points, which is reasonable in light of the actual rates, as shown in Figure 1. The solution of a differential equation is shown graphically in Figure 2.

The evaluation conservatively assumed full CEA insertion even though the rods have limited use for flux shaping, i.e. limited rod insertion. The CEAs were in place for cycles 1 through 10, which had a combined availability of 3,266 full power days, which means that the CEAs were inserted for no more than 663.0 full power days. The minimum amount of $B_{10}$ remaining in the CEAs at 663.0 full power days is 85% as shown in Figure 2.

## Criticality Analysis

The criticality analysis was performed by determining the $k_\infty$ as a function of initial enrichment and exposure, while considering the effects of calculational biases and uncertainties, perturbations such as mechanical tolerances, and postulated accident conditions to determine the minimum allowable exposure, as a function of enrichment, for an assembly with a CEA in place to be moved to Region 2. For additional conservatism, it was assumed that only 75% of the $B_{10}$ calculated value remains in the CEAs to be used in Region 2.

## Fuel Burnup Uncertainty

The uncertainty caused by the fuel burnup is estimated to be no more than 5% of the calculated reactivity loss from the fresh state to the required exposure. This is based on calculated versus experimental results for fuel lifetime, on the ability of LEOPARD to accurately track fuel depletion, and on the ability of CINDER to predict radionuclide concentration.

## Perturbations

The perturbations considered in the original Fort Calhoun Nuclear Station spent fuel rack analysis included variations in fuel box dimensions, water channel width, stainless steel box thickness, pellet density, and pool temperature. The first four have only a small effect and are considered to be of the same magnitude as in the original analysis. The most reactive temperature was 200°F in the original case, so a case with an enrichment of 4.00 w/o and 15,000 MWD/MTU exposure was run at 200°F, resulting in an increase of 0.0120 in $k_\infty$, somewhat less than in the case with no CEA. This is due to the decrease in the degree of overmoderation in the racks with the CEAs in place.

The results of the analysis are summarized in Tables 1 and 2. Table 1 shows the base case $k_\infty$ values in the Region 2 rack with CEAs inserted, and Table 2 shows the biases and uncertainties for one of the cases run, 4.00 w/o U-235 fuel at an exposure of 15,000 MWD/MTU and the adjusted $k_\infty$ values in the rack after biases and uncertainties are included.

The worst case rack $k_\infty$ value including all biases and uncertainties must be maintained below 0.95. To allow for uncertainty in determining the actual exposure and interpolating between different enrichments, the exposure required for a $k_\infty$ of 0.9450 will be determined. For an initial enrichment of 3.25 w/o this exposure is 4,900 MWD/MTU. For an initial enrichment of 4.00 w/o this exposure is 12,800 MWD/MTU. These results are shown graphically in in the proposed revision to Figure 2-10 of the Technical Specifications.

CEA Integrity

45 CEAs are currently discharged in the Spent Fuel Pool due to the CEA fingers approaching the 1% mean unrecoverable circumferential cladding strain design criteria. At the End of Cycle 9 (EOC 9) operation all full length assemblies were inspected by Combustion Engineering and a report submitted to the NRC as detailed in Reference 7. The NRC questions and the OPPD responses are contained in References 8 and 9, respectively. Reference 10 contains the NRC SER on the CEA inspection report. The inspection noted that all of the CEA fingers in the 45 CEAs had maintained structural integrity through EOC 9. Ten of 45 CEAs were discharged at this time. Because the remaining 35 CEAs, which operate through EOC 10 before discharge, were verified to have maintained their integrity at EOC 9 and calculated not to have exceeded the 1% mean unrecoverable cladding strain limit at EOC 10, it is reasonable to assume that these 35 discharged CEAs have not had a breach of cladding integrity. Thus it is concluded that all 45 of the discharged full length CEAs currently residing in the Spent Fuel Pool are considered to have maintained their integrity.

The CEAs have an estimated 85% of the absorber worth remaining which will be sufficient in the absence of a neutron flux to maintain the 0.95 $k_\infty$ requirements for the design life of the plant.

## TABLE 1

### Base Case and Adjusted k∞ Values in the Region 2 Rack with CEAs Inserted

| Initial Enrichment w/o | Fuel Exposure MWD/MTU | Base Case Computed k∞ | Adjusted* k∞ |
|---|---|---|---|
| 3.25 | 0. | 0.9599 | 0.9865 |
| 3.25 | 5000. | 0.9175 | 0.9440 |
| 3.25 | 10000. | 0.8801 | 0.9092 |
| 4.00 | 10000. | 0.9354 | 0.9629 |
| 4.00 | 15000. | 0.9023 | 0.9304 |
| 4.00 | 20000. | 0.8704 | 0.8991 |

* Adjusted for biases and uncertainties

## TABLE 2

### Summary of Reactivity Biases and Uncertainties for Fort Calhoun Region 2 Storage Rack

| Description | Reactivity Effect, k | k∞ |
|---|---|---|
| Basic Cell, 4.0 w/o U-235 @ 15,000 MWD/MTU | | 0.9023 |
| Calculational Biases | | |
| LEOPARD/PDQ/BLACKCYL model | +0.0053 | |
| Most Reactive Temperature | +0.0120 | |
| Mesh Spacing Effect | -0.005 | |
| TOTAL BIAS | +0.0168 | |
| Basic Cell Including Biases | | 0.9191 |
| Tolerances and Uncertainties | | |
| LEOPARD/PDQ/BLACKCYL model (95/95) | 0.0086 | |
| Depleted Fuel Reactivity Uncertainties | 0.0056 | |
| Minimum Assembly Pitch | 0.0035 | |
| Stainless Steel Thickness | 0.0026 | |
| Fuel Pellet Density (±0.005 in) | 0.0017 | |
| Fuel Pellet Diameter (±0.0005 in) | 0.0003 | |
| TOTAL STATISTICAL UNCERTAINTY | 0.0113 | |
| Maximum value of k∞ including all biases and uncertainties for basic cell with 4.0 w/o fuel at 15,000 MWD/MTU | | 0.9304 |

2.3 REFERENCES


1. V.E. Grob, P.W. Davison, et al., "Multi-Region Reactor Lattice Studies - Results of Critical Experiments in Loose Lattices of $UO_2$ Rods in $H_2O$", WCAP-1412, Westinghouse Atomic Power Division, dated 1960

2. Battelle Pacific Northwest Laboratories, "Critical Separation Between Subcritical Clusters of 2.35 Wt% U-235 Enriched $UO_2$ Rods in Water with Fixed Neutron Poisons", PNL-2438

3. W.L. Orr et al., "Saxton Plutonium Program, Nuclear Design of the Saxton Partial Plutonium Core", WCAP-3385-51, dated 12/65

4. R.D. Leamer et al., "$PuO_2$-$UO_2$ Fueled Critical Experiments", WCAP-3726-1, dated 7/67

5. S.R. Beirman, B.M. Durst, and E.D. Clayton, "Critical Separation Between Subcritical Clusters of 4.29 w/o U-235 Enriched $UO_2$ Rods in Water with Fixed Neutron Poisons", Battelle Pacific Northwest Laboratories, NUREG/CR-0073, dated 5/78

6. G.S. Hoovler et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel", Nuclear Technology, V51, pp 217-237, dated 12/80

7. Letter LIC-86-044, R.L. Andrews (OPPD) to A. Thadani (NRC), "CEA Inspection Results Report", dated February 28, 1986

8. Letter, D.E. Sells (NRC) to R.L. Andrews (OPPD), "CEA Inspection Results", dated July 31, 1986

9. Letter, R.L. Andrews (OPPD) to D.E. Sells (NRC), "CEA Inspection Results", dated October 3, 1986

10. Letter, W.A. Paulson (NRC) to R.L. Andrews (OPPD), "Special Report on End of Cycle 9 Control Element Assembly (CEA) Inspection Results and Impact on Cycle 10 Operation", dated March 3, 1987

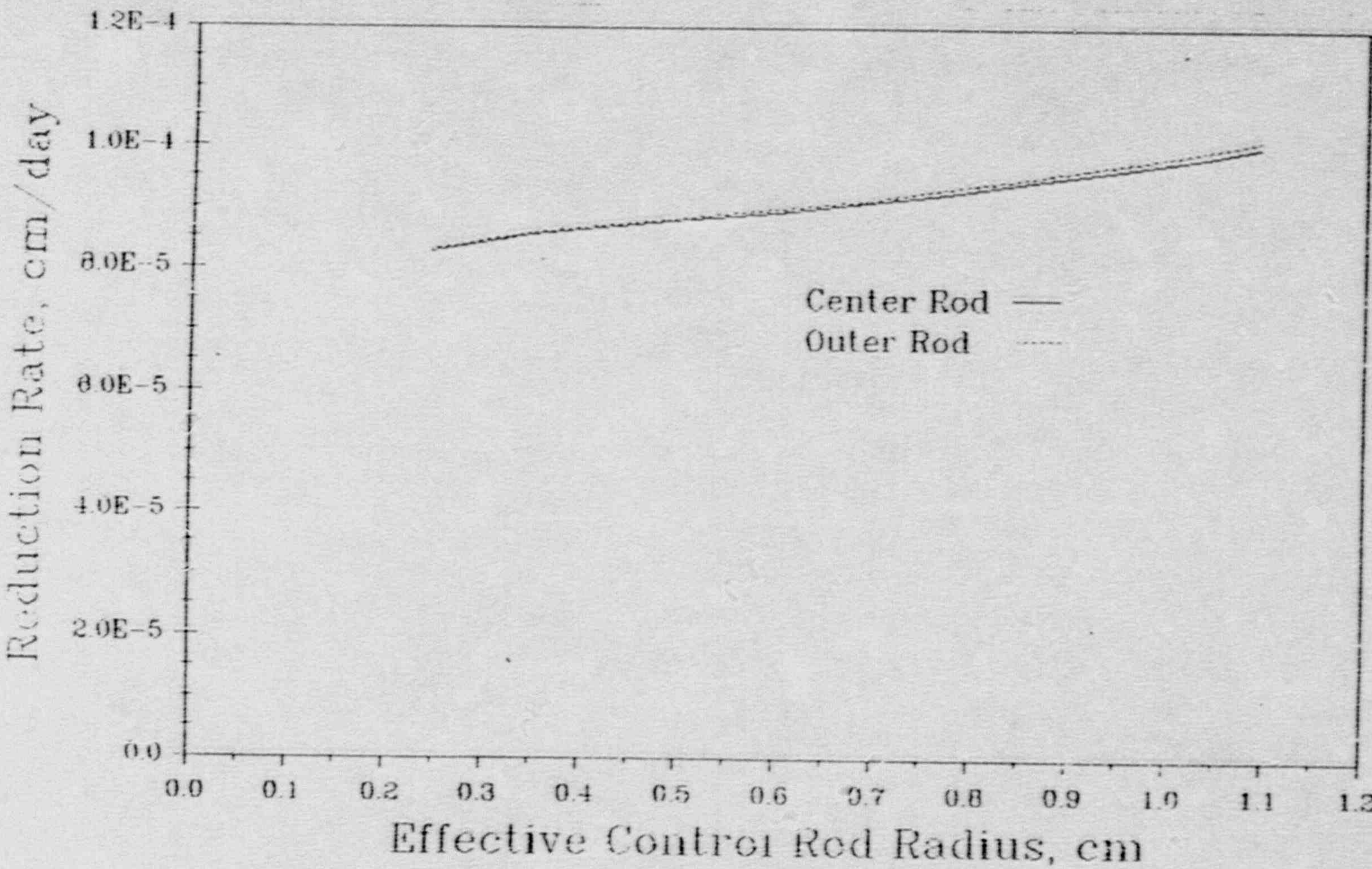



FIGURE 1 RELATION OF THE RATE OF REDUCTION OF EFFECTIVE CONTROL ROD RADIUS TO EFFECTIVE CONTROL ROD RADIUS

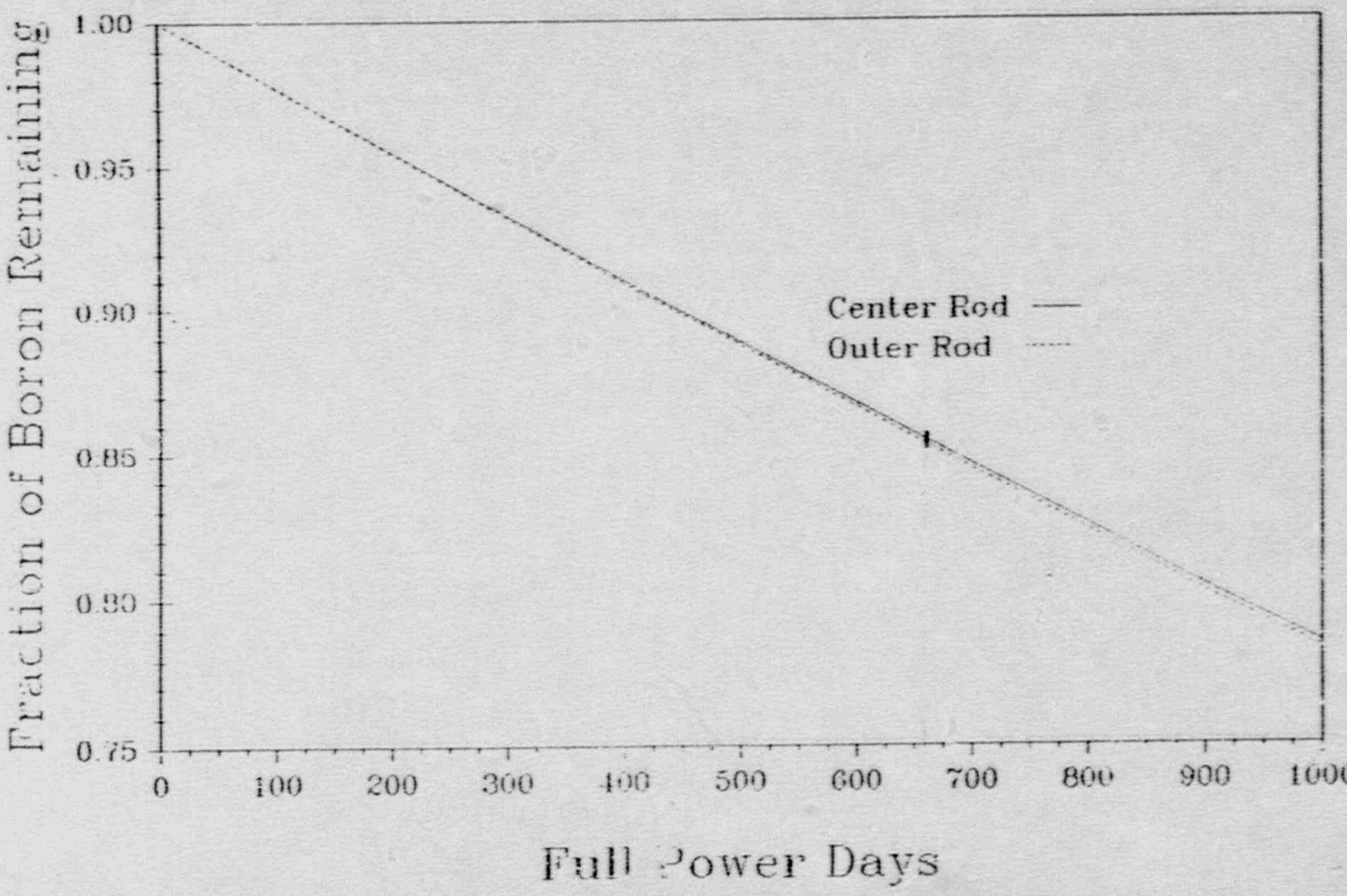



FIGURE 2 BORON REMAINING AS A FUNCTION OF THE EXPOSURE TIME AT FULL POWER

## 3.0 JUSTIFICATION

The inclusion of a discharged full length CEA in an assembly is acceptable for storage in Region 2 of the spent fuel pool since the 0.95 $k_{\infty}$ criticality criteria is met and assembly meets the burnup requirements of revised Technical Specification Figure 2-10 as a function of the initial enrichment.

Administrative controls, consisting of footnoting the requirements that the CEA remain in the fuel assembly while resident in Region 2, will assure compliance with the Technical Specification requirements. A physical restraint system as shown in Figures 3 through 6 will supplement the administrative controls. Table 3 is an installation procedure for the connection clip to be utilized as a physical restraint. Figures 3 and 4 are sketches of the installation tool while Figures 5 and 6 are sketches of the connection clip and the clip installed on the CEA/fuel assembly upper end fitting, respectively.

The material for the connecting clip was selected for compatibility with the Spent Fuel Pool environment, the CEA and the fuel assembly. The connecting clip will also be designed to support the loading of a fuel assembly and CEA assembly to facilitate any necessary movement of the pair.

Technical Data Book Figure I.B.1-3 is provided as Figure 7 to indicate potential candidates for storage in Region 2.

TABLE 3

## CONTROL ELEMENT ASSEMBLY TO FUEL ASSEMBLY CONNECTION CLIP

### INSTALLATION PROCEDURE

The Control Element Assemblies (CEA) will be attached to the fuel assemblies through use of a connection clip. The clip and tool to attach the clip are shown in Figures 1, 2, 3 and 4. The procedure to accomplish this is outlined as follows:

1. Attach top of clip (Figure 1) to bottom of tool by inserting tool bottom locking tab (Figure 2) into slot in top of clip (Figure 1) and rotating tool 90° to slot counterclockwise. This will attach clip to tool so that it can be lowered into pool.

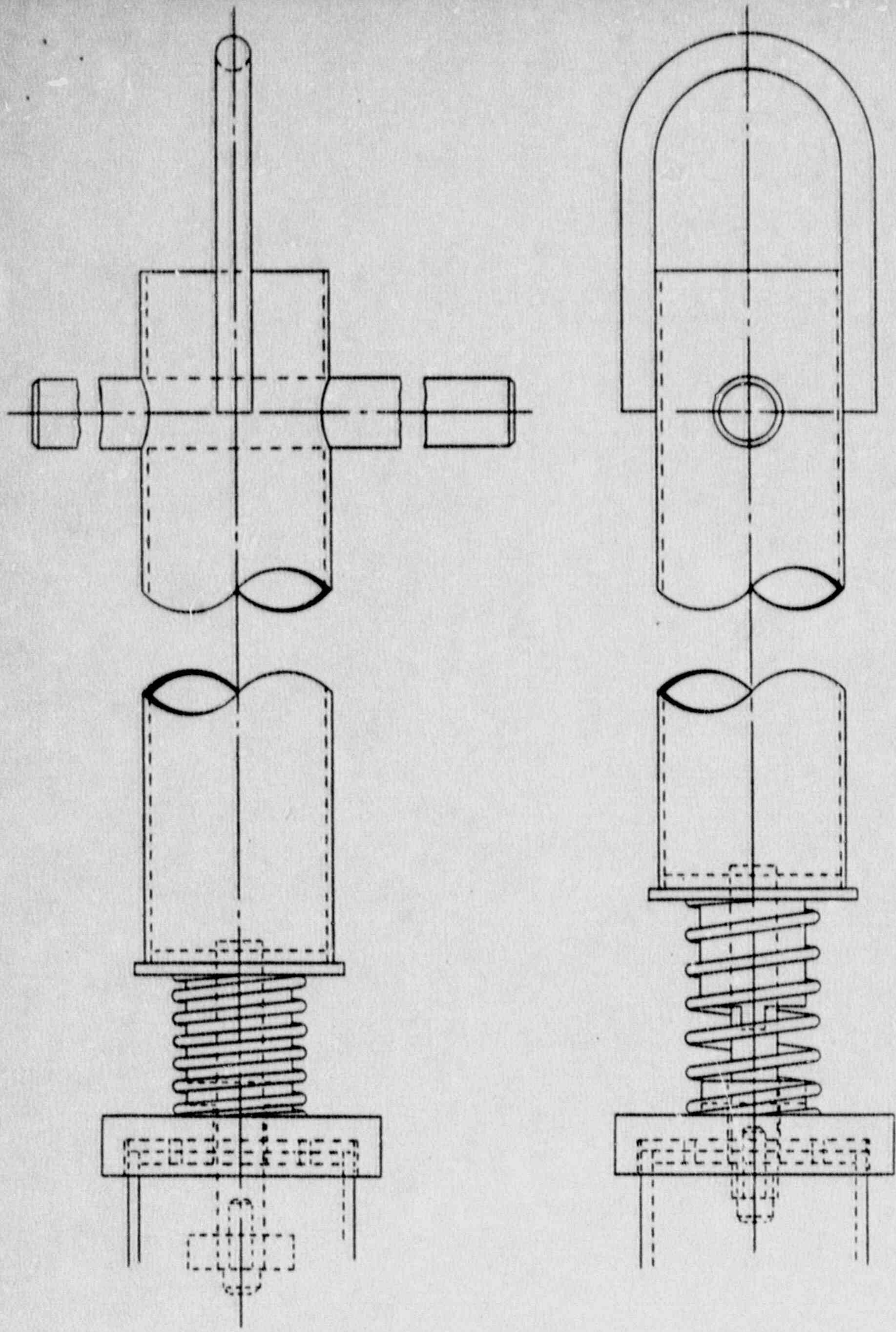
2. Lower tool with clip into pool over CEA as shown in Figure 4.

3. Press down on tool to engage rotator bar into top of slot of clip which will at the same time lower clip attachment feature below engagement bar on fuel assembly top nozzle.

4. With tool in engaged position rotate tool 90° clockwise, then release downward load on tool. The clip will now be engaged to fuel assembly top nozzle.

5. Rotate tool another 90° clockwise to align clip locking tab on bottom of tool with slot in top of clip.

6. Lift tool out of pool for next CEA to fuel assembly clip connections.



(a) (b)

FIGURE 3
TOOL FOR CONNECTION
CLIP INSERTION

Engagement position on fuel assembly

Insertion position into pool

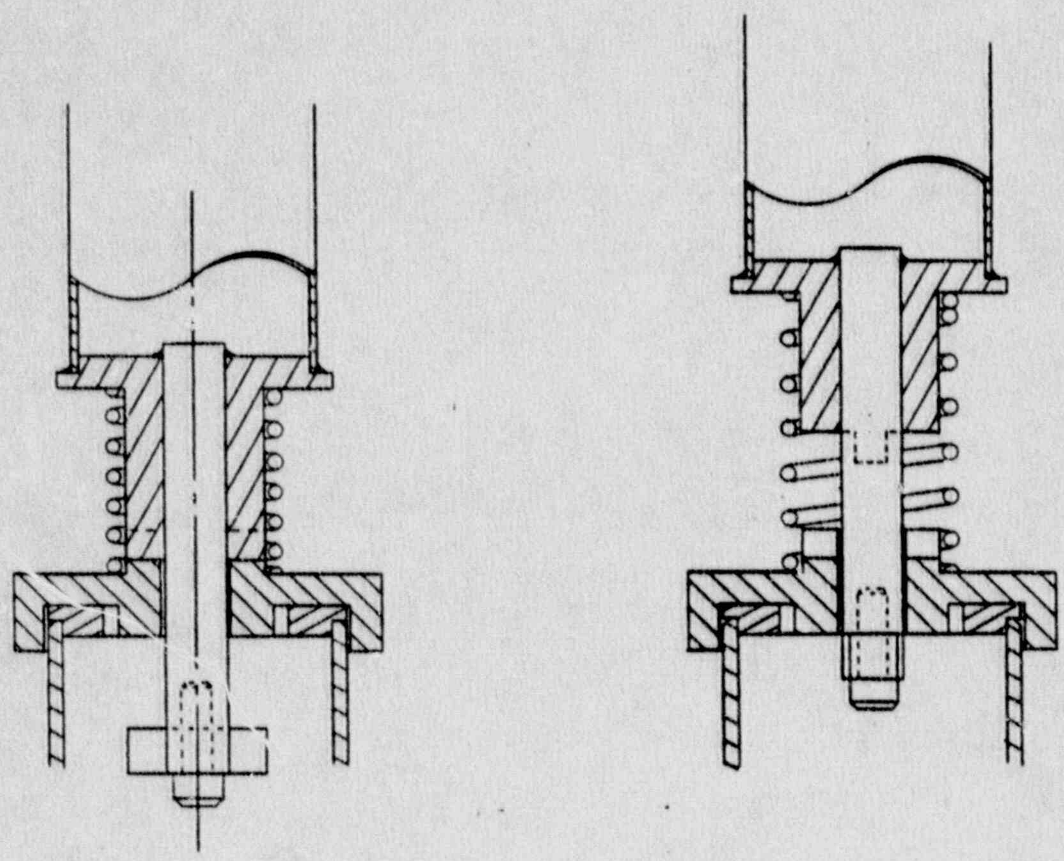


FIGURE 4

BOTTOM FEATURES
OF TOOL

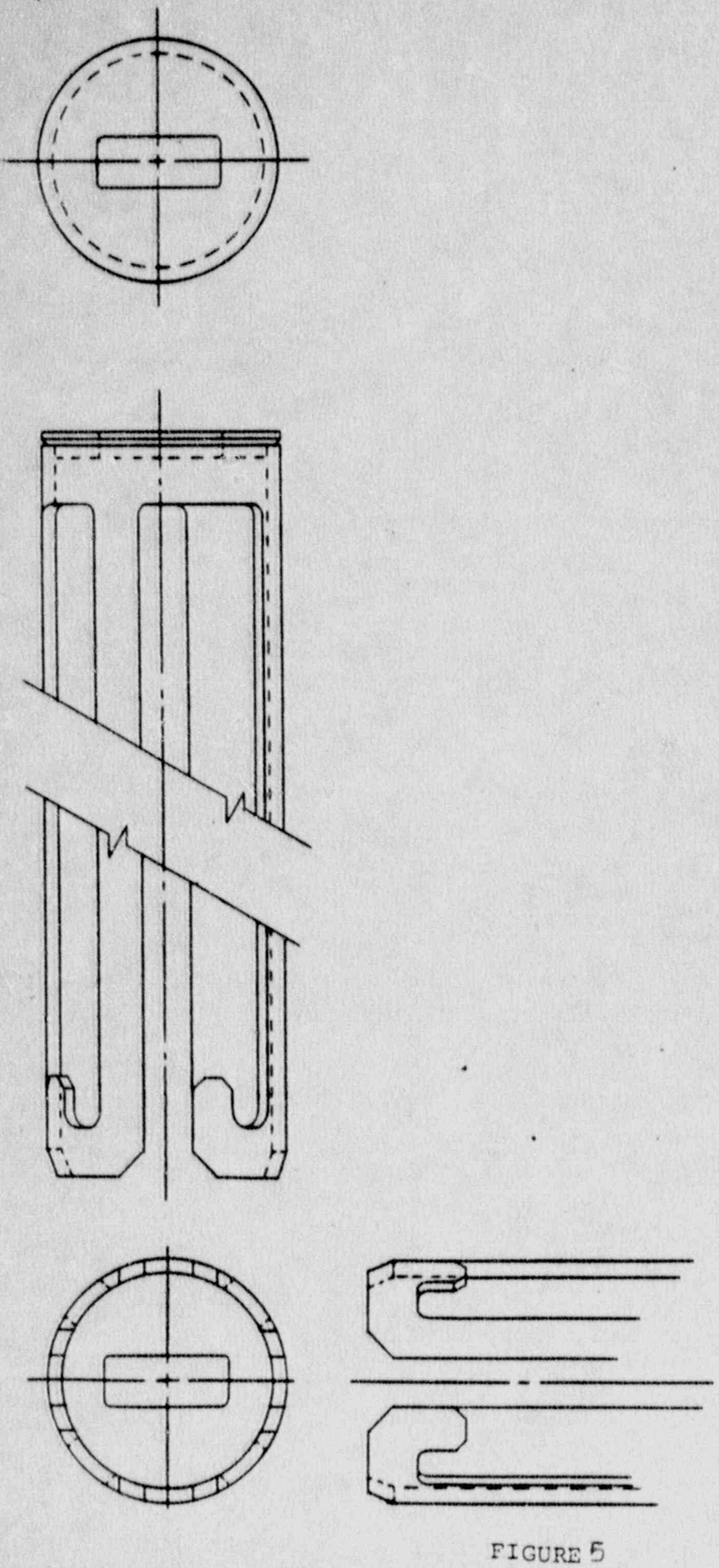
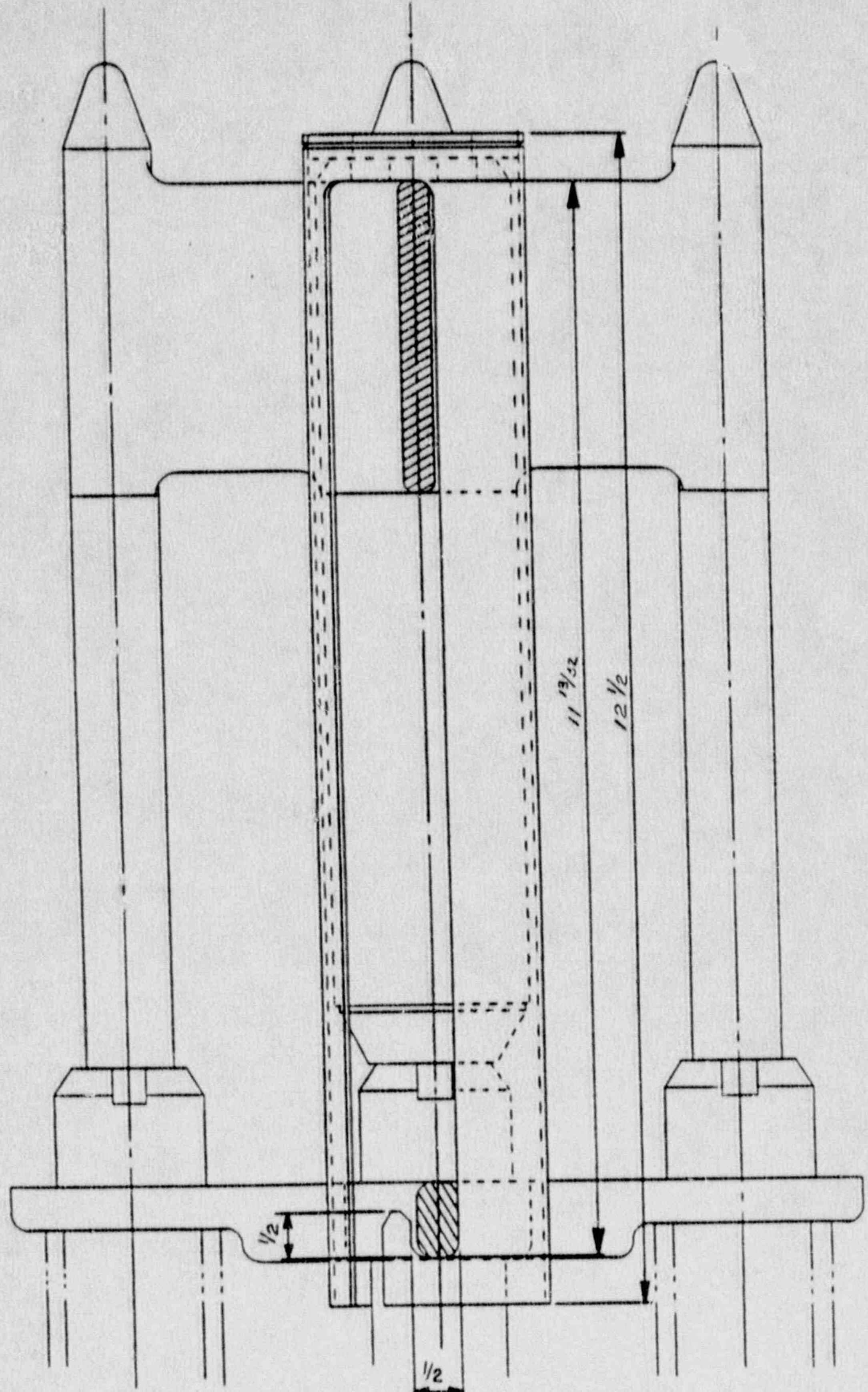


FIGURE 5
CONNECTION CLIP




SECT. 'A-A'

FIGURE 6
CONNECTION CLIP LOCKED INTO PLACE

FIGURE 7

| | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Y25 | Y26 | Y27 | ← POISONED RACKS → | | | | | | | | |
| G034 SE X25 | HA07 SE X26 | G002 SE /41 X27 | X28 | X29 | X30 | X31 | X32 | X33 | X34 | KA26 SE X35 | X36 |
| G032 SE W25 | HA05 SE W26 | D028 SE /28 W27 | W28 | W29 | W30 | HA10 SE W31 | W32 | W33 | KA14 SE W34 | SR W35 | KA27 SE W36 |
| B221 SE V25 | G011 SE V26 | G013 SE PP09 V27 | KA08 SE V28 | V29 | V30 | V31 | V32 | V33 | V34 | KA25 SE V35 | V36 |
| G019 SE U25 | B112 NW U26 | G029 SE /45 U27 | U28 | U29 | U30 | U31 | U32 | U33 | U34 | U35 | U36 |
| C110 SE T25 | HA01 SE T26 | G016 SE /15 T27 | T28 | T29 | T30 | KA19 SE T31 | T32 | T33 | T34 | T35 | T36 |
| D002 SE S25 | D032 NE S26 | D012 SE /19 S27 | KA02 SE S28 | S29 | S30 | S31 | S32 | S33 | S34 | S35 | S36 |
| D033 SE /E R25 | G020 SE PP10 R26 | C008 SE /06 R27 | R28 | R29 | R30 | R31 | R32 | R33 | R34 | R35 | R36 |
| FC04 SE Q25 | A001 SE /27 Q26 | D021 SE /04 Q27 | LA18 SE Q28 | Q29 | Q30 | KA03 SE Q31 | Q32 | Q33 | Q34 | Q35 | Q36 |
| A020 SE P25 | A023 SE /30 P26 | D020 SE /38 P27 | P28 | P29 | P30 | KA21 SE P31 | P32 | P33 | P34 | P35 | P36 |
| A008 SE N25 | A013 SE /18 N26 | D019 SE /32 N27 | N28 | N29 | N30 | N31 | N32 | N33 | N34 | N35 | N36 |
| E015 SE M25 | D018 SW M26 | C105 SW /03 M27 | M28 | M29 | M30 | KA16 SE M31 | M32 | M33 | M34 | M35 | M36 |
| E008 SE L25 | D017 NE /23 L26 | C104 SE /02 L27 | LA28 SE L28 | L29 | L30 | LA17 SE L31 | L32 | L33 | L34 | L35 | L36 |
| F007 SE K25 | D016 SE /36 K26 | C103 SE /43 K27 | K28 | K29 | K30 | K31 | K32 | K33 | K34 | K35 | K36 |
| C114 NW J25 | B103 NE J26 | B209 SE /07 J27 | J28 | J29 | J30 | J31 | J32 | J33 | J34 | J35 | J36 |
| D025 SE H25 | C116 NE H26 | B214 NW /34 H27 | H28 | H29 | H30 | KA09 SE H31 | H32 | H33 | H34 | H35 | H36 |
| A003 SE G25 | D029 SE /37 G26 | C001 NW /20 G27 | G28 | G29 | LA27 SE G30 | G31 | G32 | G33 | G34 | G35 | G36 |
| A043 SE F25 | B114 SW /C F26 | D027 SE /13 F27 | F28 | F29 | KA32 SE F30 | F31 | F32 | F33 | F34 | F35 | F36 |
| C115 SE E25 | B113 SW /44 E26 | B110 NE /08 E27 | E28 | E29 | E30 | E31 | E32 | LA44 SE E33 | E34 | LA41 SE E35 | LA40 SE E36 |
| D026 SE D25 | B009 SE /A D26 | B204 NE /10 D27 | D28 | D29 | D30 | LA19 SE D31 | D32 | LA37 SE D33 | LA35 SE D34 | LA42 SE D35 | LA39 SE D36 |
| B124 SW C25 | C26 | G023 SE PP08 C27 | FPB C28 | IA07 SE C29 | IA08 SE C30 | HA14 SE C31 | IA21 SE C32 | IA22 SE C33 | IA23 SE C34 | IA24 SE C35 | KA18 SE C36 |
| G043 SE PP05 B25 | B26 | B27 | T CAN B28 | IA01 SE B29 | IA02 SE B30 | LA38 SE B31 | LA43 SE B32 | IA04 SE B33 | IA05 SE B34 | IA06 SE B35 | C003 NW /D B36 |
| F.C.2 A25 | F.C.1 A26 | IC CAN /SP A27 | SC CAN A28 | C004 NW PD A29 | C012 NW /24 A30 | C007 SE /21 A31 | C006 SE /16 A32 | C016 SE /25 A33 | C005 SE /11 A34 | LA36 SE A35 | C002 NW /B A36 |

**SP** = SPOOL PIECE
**IC CAN** = INCORE CAN
**FPB** = FUEL PIN BASKET
**T CAN** = TRASH CAN
**SC CAN** = SURVEILANCE CAP. CAN
**PD** = PLPR DUMMY
**PP** = PLPR
**CED** = C E DUMMY
**SR** = SAMPLE RACK
**F.C.1 & 2** = FILTER CANNISTER 1 & 2

○ potential candidates for Region 2 storage.

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## NO SIGNIFICANT HAZARDS

This proposed amendment to the Technical Specifications will allow spent fuel assemblies with limited exposure to be stored in Region 2 of the spent fuel pool, if a full length Control Element Assembly (CEA) is inserted in the fuel assembly prior to the move to Region 2.

The Technical Specification document changes are defined by the updated Figure 2-10 of Section 2.8.

Four postulated accident situations are of potential concern from a criticality standpoint. These are a boron dilution accident, a dropped fuel assembly lodging sideways on top of the rack, a loss of pool water, and failure to insert or inadvertently remove a CEA from a low burnup spent fuel assembly. The postulated boron dilution accident is analyzed as the base case since the base case includes no soluble boron in the pool.

Since the rack design ensures enough separation between the top of the fuel and the top of the rack (about 27") to effectively decouple the dropped assembly from the rack, a postulated dropped assembly will have no effect on reactivity. Also, the effect of the dropped assembly is to reduce neutron leakage, and no credit for axial leakage was taken in the base case.

A postulated loss of pool water reduces the effective water density of the pool if the loss of water is due to the pool boiling, and can lead to an increase in reactivity if the pool is initially overmoderated, as it is in Region 2. However, since this is an accident, only the consideration of each single failure in isolation is required. This means that credit may be taken for the minimum soluble boron concentration of 1700 ppm when analyzing this accident. Note that 1700 ppm is less than the proposed Cycle 13 value of 1900 ppm and therefore is conservative. This level of boron more than compensates for the increase in k due to a reduction in effective water density in the original spent fuel rack storage analysis. The insertion of a CEA displaces water reducing the initial level of overmoderation; thus, the previous analysis conservatively bounds this case.

A low burnup spent fuel assembly requiring an inserted CEA for transport and storage into Region 2 will be administratively controlled to preclude a criticality accident. This will be effected by attaching a clip to tie the CEA and fuel assembly together in Region 1 prior to transfer to Region 2. The clip cannot be removed by the grapple on the fuel handling machine. The clip will prevent the inadvertent removal of the CEA from the fuel assembly.

## BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change allows the storage of fuel not meeting the burnup of criteria of the current Technical Specification to be stored in Region 2 of the spent fuel pool provided a CEA is inserted and the assembly burnup meets the acceptance criteria of the proposed revised

Technical Specification Figure 2-10. The Region 2 racks have been analyzed to ensure the minimum 0.95 $K_\infty$ margin is maintained. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the same Technical Specification controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Specifications involving the storage of spent fuel in Region 2 of the storage pool conform to current plant conditions and, therefore, preserve the margin of safety. Storage procedures contain strict administrative requirements and burnup requirements which are independently verified as acceptable and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

ITEM 4

ALLOW DIRECT TRANSFER OF SPENT FUEL
FROM THE REACTOR CORE TO THE
SPENT FUEL POOL REGION 2

## DISCUSSION OF CHANGE

The proposed amendment to the Technical Specifications will specifically allow spent fuel transfer directly from the reactor core to the spent fuel pool Region 2.

Prior to the transfer of a fuel assembly during refueling operations, the Special Procedure SP-BURNUP-1 must be completed to insure an independent review of the assembly burnup to verify all variables are within the acceptable ranges identified in Technical Specification Figure 2-10.

The direct transfer of fuel from the reactor core to the spent fuel pool Region 2 requires adequate procedures and independent checks to ensure that the spent fuel meets the acceptance criteria defined in Technical Specification Figure 2-10. The existing procedures are identified and demonstrate that adequate control is provided.

The analysis required to directly transfer fuel from the reactor core to the spent fuel pool Region 2 requires a demonstration that adequate procedures and administrative controls are in place to insure that the spent fuel meets the acceptance criteria defined in Technical Specification Figure 2-10 prior to the transfer.

The Fort Calhoun Station Unit No. 1 Operating Procedure OP-11, "Reactor Core Refueling Procedure" defines the steps required to provide a safe and organized method of refueling the reactor. This procedure covers the fuel movement sequence (Appendix A), and post core loading verification checks. The Low Power Physics Testing (SP-PRCPT-1) procedure is an additional check that ensures the core is loaded as planned and analyzed.

The transfer of fuel from the reactor vessel to the spent fuel pool is covered by the same procedure (OP-11) along with the fuel movement sequence. In this case the fuel burnup determination, performed using Special Procedure SP-BURNUP-1, must be completed prior to fuel movement into Region 2 of the spent fuel pool. This procedure is used to determine and verify the burnup of each fuel assembly.

The administrative procedures covering the movement of fuel are considered to adequately ensure that the fuel will meet the acceptance criteria for burnup prior to movement to Region 2 of the spent fuel pool. Furthermore, the potential risk of dropping or mislocating a fuel assembly with a two-step operation is minimized.

## JUSTIFICATION

The direct transfer of spent fuel from the reactor vessel to the spent fuel pool Region 2 allows for safe storage on the basis that all administrative procedures are followed. Furthermore, the potential risk of dropping or mislocating a fuel assembly with the additional handling required in a two-step operation is minimized.

## NO SIGNIFICANT HAZARDS

The proposed amendment to the Technical Specifications will specifically allow spent fuel transfer directly from the reactor core to the spent fuel pool Region 2.

The Technical Specification changes are defined on page 2-38 of Section 2.8.

Prior to the transfer of a fuel assembly during refueling operations, the Special Procedure SP-BURNUP-1 must be complete to insure an independent review of the assembly burnup and to verify the burnup is within the acceptable ranges identified in Technical Specification Figure 2-10.

## BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change allows the direct transfer of fuel from the reactor core to Region 2 of the spent fuel pool provided all procedures have been completed and the assembly burnup meets the acceptance criteria with no changes in administrative specifications. The final storage location of any given fuel element will be the same using a one-step or a two-step transfer process. Therefore, this change does not significantly increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the existing Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Administrative specifications involving the transfer of spent fuel to the storage pool conform to current plant conditions and, therefore, preserve the margin of safety. Changes in the fuel transfer procedures contain strict administrative procedures and ensure that the burnup requirements are verified as acceptable and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

ITEM 5

DELETE REQUIREMENT FOR FUEL
PERFORMANCE REPORT AT END
OF EACH CYCLE

## DISCUSSION OF CHANGE

This request is an administrative change to the Fort Calhoun Station Technical Specifications previously requested by OPPD and approved in Amendment No. 77 to Facility Operating License No. DPR-40.

The requested Technical Specification revision is administrative because of the previously approved request documented in Amendment No. 77 (Reference 1). The additional reporting was performed as part of high fuel burnup demonstration project that was completed, hence the comprehensive fuel inspections and reporting are no longer required.

The analysis discussions are contained in the reference documents identified herein. Copies of the appropriate pages are furnished for your convenience.

REFERENCES

1. U. S. NRC letter from E. G. Tourigny to W. C. Jones (OPPD), dated 4/26/84 issuing Amendment No. 77

NOTE: Copies of selected pages from the above reference are attached.

4/27/84 - WEM, GPB, WCJ, MOG, KJM, RLJ, WGG, TJM, JKG, DDW, PMS, FAT, FFF, JJF (3), DJM, TRR, RRM, FCFile

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 26 1984

84-120

Docket No. 50-285

Mr. W. C. Jones
Division Manager, Production
Operations
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 77 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated February 14, 1984, as supplemented by letters dated March 28, April 4 and 16, 1984.

The amendment authorizes changes to the Fort Calhoun Station, Unit No. 1 Technical Specifications which are required to support the operation of the unit at full rated power during Cycle 9. Specifically, the following specifications are changed: minimum departure from nucleate boiling (DNB) ratio, total unrodded planar radial peak, unrodded integrated total radial peak, lower bound of the moderator temperature coefficient and linear heat rate measurement-calculation uncertainty factor. Because these specifications are changed, the following figures are changed: Thermal Margin/Low Pressure Safety Limit, Thermal Margin/Low Pressure Limiting Safety System Settings, Limiting Condition for Operation for Excore Monitoring and Linear Heat Rate, Limiting Condition for Operation for DNB Monitoring, and $F_R^T$ and $F_{xy}^T$ and Core Power Limitations. A steam generator differential pressure reactor protective system trip is added to the technical specifications. Instrument Operating Requirements and Minimum Frequencies for Checks, Calibration, and Testing are added for this new trip.

The amendment also revises the Technical Specifications for the reactor coolant system pressure-temperature limits for operation to 8.5 effective full power years. The current technical specifications permit operations to 7.0 effective full power years, which will occur during Cycle 9 operation. This necessitates a change in the following figure: Heatup-Reactor Not Critical, Cooldown-Reactor Not Critical, and Predicted Radiation Induced NDTT Shift. Last, the amendment deletes the spent fuel inspection requirements.

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

E. G. Tourigny, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosure:
1. Amendment No. 77 to DPR-40
2. Safety Evaluation

cc w/enclosure:
See next page

Omaha Public Power District

cc:

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The ASGTPTF can be tested with the reactor in operation or shut down. The proposed Technical Specification changes relating to the ASGTPTF modification include provision for monthly testing of the SG pressure instrument channels (i.e., a bistable trip test), a channel check to be performed each shift, and calibration of the SG pressure sensors during each refueling outage. Similar Technical Specification surveillance requirements currently existing for the SG low pressure RPS instrumentation will remain unchanged.

Based on our review of the licensee's submittal, we conclude that the electrical, instrumentation and control aspects of the proposed ASGTPTF modifications to the Reactor Protection System and the associated proposed Technical Specifications comply with the applicable criteria of Section 7.2 "Reactor Trip System" of the Standard Review Plan (NUREG-0800), and therefore, are acceptable.

## 8.0 DELETION OF SPENT FUEL INSPECTION REQUIREMENTS

OPPD has requested that Section 5.9.3.h of the Fort Calhoun Technical Specifications be eliminated. This specification requires a report on fuel performance following each refueling outage. A comprehensive fuel assembly demonstration project has recently been completed by OPPD which resulted in the conclusion that no rod perforations nor anomalies existed in a lead demonstration assembly after six cycles (approximately 56 GWD/MTU) of irradiation.

Based on this and on the discussion presented in Reference 16, we conclude that the elimination of this requirement involves no significant hazards considerations in accordance with the criteria contained in 10CFR50.92. Reference 16 points out that we currently do not require reporting for routine surveillance of standard fuel designs. However, it refers to the Licensee Event Report System Rule (10CFR50.73) which became effective this year for guidance as to when reports should be issued. The supporting document for this rule states that reportable situations include "fuel cladding failures in the reactor or in the storage pool that exceed expected values, that are unique or widespread, or that resulted from unexpected factors" (NUREG-1022).

## 9.0 CONCLUSIONS - PHYSICS, THERMAL-HYDRAULICS, TRANSIENTS AND FUELS

The staff has reviewed the information presented in the Fort Calhoun Cycle 9 reload report and in OPPD responses to our requests for additional information. We find the proposed reload and the associated modified Technical Specifications acceptable.

## JUSTIFICATION

The requested Technical Specification revisions should be provided as previously requested and approved.

NO SIGNIFICANT HAZARDS

The proposed Technical Specification change deletes the requirement to provide a fuel performance report after each cycle of operation.

The Technical Specification document change is defined on page 5-15 of Section 5.9.

The proposed change for deleting Technical Specification requirement 5.9.3.h has been previously approved in the Safety Evaluation Report for Amendment No. 77 to the facility operating license. Thus, this change is considered administrative and does not affect the safe operation of the plant. The requirement to provide the fuel performance report is duplicate to the Licensee Event Report System Rule (10CFR50.73) to report unique, widespread or unexpected fuel cladding failures in the reactor or spent fuel pool.

BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve an increase in the probability or consequence of an accident previously evaluated. This change implements a previously approved revision to the Technical Specifications. Therefore, this change does not increase the probability or consequences of a previously evaluated accident. This change is considered administrative.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the current Technical Specification administrative controls prevents the possibility of anew or different kind of accident.

3) Involve a significant reduction in a margin of safety. Administrative specifications involving the deletion of the fuel performance report will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.