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Docket Nos. 50-325/324

MEMORANDUM FOR: Thomas M. Novak, Assistant Director
for Operating Reactors, DL

FROM: William E. Kreger, Assistant Director
for Radiation Protection, DSI

SUBJECT: BRUNSWICK, UNIT NOS. 1 AND 2, SPENT FUEL POOL EXPANSION (TAC #43797)

In accordance with TAC #43797, the Effluent Treatment Systems Branch (ETSB) has completed the review and evaluation of the April 16, 1980 letter from the licensee, Carolina Power and Light Company (CP&L) which included a document entitled "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Spent Fuel Storage Expansion Report" and which provided information on the proposed expansion of the storage capacity of the spent fuel pool (SFP) for amending technical specification 5.6 of DPL-71 and DPL-62. The review was performed by J. S. Boegli, ETSB (Ext. 27634).

The present licenses for Brunswick, Unit Nos. 1 and 2, permit a spent fuel storage capacity of 1386 BWR fuel assemblies in each SFP. In addition, the technical specifications for each license permit space to store PWR fuel assemblies from CP&L's H. B. Robinson Plant. There are presently 304 PWR fuel assemblies at the Brunswick Plant and no additional space is expected. This modification proposes to increase the licensed storage capacity of the SFP to 1803 BWR and 160 PWR fuel assemblies at Unit 1, and 1839 BWR and 144 PWR fuel assemblies at Unit 2.

Enclosure 1 is suitable for inclusion in the Safety Evaluation. Enclosure 2 is suitable for inclusion in the Environmental Impact Appraisal.

Or[illegible]d by:
Willi[illegible]. Kreger

William E. Kreger, Assistant Director
for Radiation Protection
Division of Systems Integration

Enclosures:
1. Safety Evaluation Input
2. Environmental Impact Appraisal Input
3. Calculation Sheets (2)

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DATE	11/23/81	11/23/81	11/27/81	11/27/81			

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Enclosure 1

SAFETY EVALUATION INPUT FROM
THE EFFLUENT TREATMENT SYSTEMS BRANCH
IN THE MATTER OF THE BRUNSWICK, UNIT NOS. 1 AND 2
SPENT FUEL POOL EXPANSION APPLICATION

3.5.1.3 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation, dated November 1973. There will be no change in the waste treatment system or in the conclusions given in Sections 9.0 and 11.0 of the evaluation of these systems because of the proposed modification. Our evaluation of the SFP cleanup system, in light of the proposed modification, has concluded that any resultant additional burden on the system is minimal and therefore the existing SFP cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water within acceptably low levels.

3.5.2 Conclusions

Our evaluation of the radiological considerations supports the conclusion that the proposed modification to the spent fuel pool at Brunswick, Unit Nos. 1 and 2, is acceptable because:

(1) The conclusions of the evaluation of the waste treatment systems, as found in the Brunswick, Unit Nos. 1 and 2, Safety Evaluation Report (November 1973), are unchanged by the modification of the SFP.

(2) The existing SFP cleanup system is adequate for the proposed modification.

Enclosure 2

ENVIRONMENTAL IMPACT APPRAISAL INPUT FROM
THE EFFLUENT TREATMENT SYSTEMS BRANCH
IN THE MATTER OF THE BRUNSWICK, UNIT NOS. 1 AND 2
SPENT FUEL POOL EXPANSION APPLICATION

1.3 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statement (FES) dated January 1974. There will be no change in the waste treatment systems described in Section III.D.2 of the FES because of the proposed modification.

1.4 Spent Fuel Pool Cleanup System

The SFP cleanup system is part of the pool cooling system. It consists of a demineralizer with inlet and outlet filters, and the required piping, valves, and instrumentation. There is also a separate skimmer system to remove surface dust and debris from the SFP. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at acceptably low levels.

We expect only a small increase in radioactivity released to the pool water as a result of the proposed modification, as discussed in Section 2.2.1, and we therefore conclude the spent fuel pool cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

2.2.2 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional fuel assemblies for a longer period of time would be the noble gas radionuclide Krypton-85 (Kr-85). As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no longer a significant release of fission products, including Kr-85, from stored fuel containing cladding defects. One hundred forty (140) fuel assemblies are expected to be stored following each March refueling at Unit 1 and each November refueling at Unit 2. Since space must be reserved to accommodate a complete reactor core unloading operation (nominally 560 fuel assemblies), and module spaces are reserved for PWR fuel assemblies, the useful pool capacity is 1243 fuel assemblies at Unit 1 and 1279 fuel assemblies at Unit 2. At an input of 140 fuel assemblies per year, the storage capacity is approximately 9 years at each unit.

For the simplest case, we assumed that all of the Kr-85 that is going to leak from defected fuel is going to do so in the 12 month interval between refuelings. In other words, all of the Kr-85 available for release is assumed to come out of the fuel before the next batch of fuel enters the pool. Our calculations show that the expected release of Kr-85 from a 140 fuel assembly refueling is approximately 79 ~~62~~ Ci each 12 months. As far as potential dose to offsite populations is

2.2.2 concerned, this is actually the worst case, since each refueling would generate a new batch of Kr-85 to be released. As more and more fuel is added to the pool, one might think that this would increase the releases, but according to the terms of our model, this is not the case since all of the Kr-85 available for release has already left the defected fuel previously stored in the pool before the next batch enters, with the result that the annual releases are not cumulative but remain approximately the same. In other words, the enlarged capacity of the pool has no effect on the total amount of Kr-85 released to the atmosphere each year. Thus, we conclude that the proposed modifications will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above the 150 F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FES. Most airborne releases of tritium

2.2.2 and iodine result from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than the spent fuel pool. Therefore, even if there were a higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increased stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If it is desired to reduce levels of radioiodine, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. In addition, the station radiological effluent Technical Specifications which are not being changed by this action, limit the total releases of gaseous activity.

2.2.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity in the SFP water to acceptable levels.

2.2.3 The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two resin beds a year due to the increased operation of the spent fuel pool cleanup system. The annual average volume, per unit, of solid wastes shipped from the Brunswick Plant during 1978 through 1980 was 15,000 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 160 cubic feet per unit per year, the increase in total waste volume shipped would be approximately 1% and would not have any significant additional environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification are contaminated and will be disposed of as low level solid waste. We have estimated that approximately 7000 cubic feet of solid radwaste will be removed from the plant because of the proposed modification. Averaged over the lifetime of the plant this would increase the total waste volume shipped from the facility by less than 3%. This will not have any significant additional environmental impact.

2.2.4 Radioactive Material Relesed to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. Since the SFP cooling and cleanup system operates as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of this modification. The SFP demineralizer resin removes soluble radioactive material from the SFP water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins. If any radioactive material is transferred from the spent resin to the sluice water, it will be removed by the liquid radwaste system for processing. After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

SFP Modification
Estimate Release Rate of Kr-85

Data Brunswick, Unit 1 Docket No. 50-325 DRR-71 2436 MWt

Core = 560 fuel assemblies

Single Refueling = 140 core assemblies / year

Cladding = Zircaloy-4

Burnup = 35,000 MWd/MT

Weight of UO_2 in Core = 100 MT

Escape rate Coeff. of Kr-85 = 6.5×10^{-8} sec

Fission Yield of Kr-85 = 0.0034

Present Capacity = $\frac{1386 - 560}{140}$ = 5.9 years

Future Capacity = $\frac{1803 - 560}{140}$ = 8.9 years

Failed Fuel Fraction (NUREG-0017) = .0012

Half-life (Kr-85) = 10.7 years

Eff. Full Power Days = (1168 days)
(3.2 years)
(.80 availability)

$$\text{Amt Kr-85 in fuel} \leq \frac{\text{Production}}{\lambda_{decay} + \lambda_{leakage}}$$

$$\text{Production} = \frac{\overset{\text{atoms/f}}{0.0034} \times \overset{\text{f/MWsec}}{3.12 \times 10^{16}} \times \overset{\text{MWD/Mt}}{35,000}}{1168 \text{ days}}$$

= 3.18×10^{15} atoms/Mtsec

($\lambda_{decay} = 2.05 \times 10^{-9}$/sec, $\lambda_{leak} = 6.5 \times 10^{-8}$/sec)

Amt Kr-85 in fuel $\leq$ 4.74×10^{22} atoms/Mt

$\leq$ 2645 Curies/Mt

This model assumes that all Kr-85 in the failed fuel assemblies will be released before the spent fuel is removed from the pool. The Kr-85 release rate is assumed constant with time. The additional capacity allows spent fuel to remain in the pool up to 8.9 years. Neglecting decay and assuming all spent fuel has failed, Kr-85 release rate is

$$\text{Kr-85}\ \frac{\text{Ci/yr}}{\text{Mt}} = \frac{2645}{8.9}$$

$$\text{Kr-85}\ \left(\frac{\text{Ci/yr}}{\text{Mt}}\right) = 297\ \text{Ci/yr/Mt}$$

The failed fuel fraction is 0.0012 cladding. The weight of a single fuel assembly in UO_2 is 0.178 Mt. The number of fuel assemblies stored each year is 140. The ~~additional~~ capacity of the pool because of the expansion is 1243 assemblies. The ~~additional~~ Kr-85 release rate with the pool full and no decay is:

R(Ci/yr)(Kr-85) = 297 Ci/yr/Mt x 0.178 Mt/assemb x 1243 assemb x .0012

R = 78.9 Ci/yr

This release rate is conservative because:

1. radioactive decay was neglected;
2. release rate of Kr-85 from failed fuel should be exponentially decreasing - the release rate is dependent on the amount of Kr-85 within the fuel;
3. release rate of Kr-85 should decrease as the spent fuel cools; and
4. this release rate assumes the pool is always full.

Since 140 assemblies will be added each refuelling and assuming one refuelling each year, the increment in the Kr-85 release rate each year, until the pool is full, is:

78.9 Ci/yr x 140/1243 = 8.9 Ci/yr/refuelling

If the decay of Kr-85 is accounted for at the end of each year after refuelling, the release rate when the pool is full is:

$$8.9\ \frac{\text{Ci}}{\text{yr}} \sum_{n=1}^{9} e^{-(9 - n)\frac{\ln 2}{10.7\text{yr}}} = 62.7\ \text{Ci/yr}$$

per refuelling

SFP Modification
Estimate Release Rate of Kr-85

Data Brunswick, Unit 2 Docket No. 50-324 DPR-62

Core = 560 fuel assemblies

Single Refueling = 140 core assemblies

Cladding = Zircaloy-4

Burnup = 35,000 MWd/MT

Weight of UO_2 in Core = 100 MT

Escape rate Coeff. of Kr-85 = 6.5×10^{-8} sec

Fission Yield of Kr-85 = 0.0034

Present Capacity = $\frac{1386-560}{140}$ = 5.9 years

Future Capacity = $\frac{1839-560}{140}$ = 9.1 years

Failed Fuel Fraction (NUREG-0017) = .0012

Half-life (Kr-85) = 10.7 years

Eff. Full Power Days = (1168 days) (3.2 years) (.80 availability)

$\frac{560}{140}(365)(0.8) = 1168$ d

$\frac{560}{140}(0.8) = 3.2$ yr

Amt Kr-85 in fuel $\leq \dfrac{\text{Production}}{\lambda_{decay} + \lambda_{leakage}}$

$$\text{Production} = \frac{\overset{\text{atoms/f}}{0.0034} \times \overset{\text{f/MWsec}}{3.12 \times 10^{16}} \times \overset{\text{MWD/Mt}}{}}{1168 \text{ days}} =$$

= 3.18×10^{15} atoms/Mtsec

($\lambda_{decay} = 2.05 \times 10^{-9}$/sec, $\lambda_{leak} = 6.5 \times 10^{-8}$/sec)

$$\frac{4.74(10^{22})(85)(395)}{6.02(10^{23})} =$$

Amt Kr-85 in fuel $\leq$ 4.74×10^{22} atoms/Mt

$\leq$ 2645 Curies/Mt

This model assumes that all Kr-85 in the failed fuel assemblies will be released before the spent fuel is removed from the pool. The Kr-85 release rate is assumed constant with time. The additional capacity allows spent fuel to remain in the pool up to 9.1 years. Neglecting decay and assuming all spent fuel has failed, Kr-85 release rate is

$$\text{Kr-85}\ \frac{\text{Ci/yr}}{\text{Mt}} = \frac{2645}{9.1}$$

$$\text{Kr-85}\ \left(\frac{\text{Ci/yr}}{\text{Mt}}\right) = \underline{\quad 291 \quad}\ \text{Ci/yr/Mt}$$

The failed fuel fraction is 0.0012 cladding. The weight of a single fuel assembly in UO_2 is 0.178 Mt. The number of fuel assemblies stored each year is 140. The ~~additional~~ capacity of the pool because of the expansion is 1279 assemblies. The ~~additional~~ Kr-85 release rate with the pool full and no decay is:

R(Ci/yr)(Kr-85) = 291 Ci/yr/Mt x 0.178 Mt/assemb x 1279 assemb x .0012

R = 79.5 Ci/yr

This release rate is conservative because:

1. radioactive decay was neglected;
2. release rate of Kr-85 from failed fuel should be exponentially decreasing - the release rate is dependent on the amount of Kr-85 within the fuel;
3. release rate of Kr-85 should decrease as the spent fuel cools; and
4. this release rate assumes the pool is always full.

Since 140 assemblies will be added each refuelling and assuming one refuelling each year, the increment in the Kr-85 release rate each year, until the pool is full, is:

79.5 Ci/yr x 140/1279 = 8.7 Ci/yr/refuelling

If the decay of Kr-85 is accounted for at the end of each year after refuelling, the release rate when the pool is full is:

$$8.7\ \frac{\text{Ci}}{\text{yr}} \sum_{n=1}^{9} e^{-(9 - n)\frac{\ln 2}{10.7\text{yr}}} = \underline{\quad 61.3 \quad}\ \text{Ci/yr}$$

per refuelling

December 1, 1981

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE PNO-II-81-112

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by IE staff on this date.

FACILITY: Carolina Power and Light Company
H. B. Robinson
Docket No. 50-261
Hartsville, South Carolina

Licensee Emergency Classification:
X Notification of Unusual Event
___ Alert
___ Site Area Emergency
___ General Emergency
___ Not Applicable

SUBJECT: UNUSUAL EVENT AT H. B. ROBINSON UNIT 2

An unusual event was declared on November 30 at H. B. Robinson Unit 2 when, at 1:44 p.m. (EST), with the plant in hot shutdown for reactor coolant system integrity tests prior to startup, a leaking valve gasket in the reactor coolant pump (RCP) seal injection line, located in the charging pump room, resulted in the spilling of 1,500 gallons of primary coolant water to the auxiliary building floor.

Precautionary evacuation of the auxiliary building was conducted. No overexposures, environmental release, or significant personnel contamination occurred.

Termination of RCP seal injection flow required the shutdown of the reactor coolant pumps. This terminated pressurizer spray, resulting in a reactor coolant system (RCS) pressure increase.

Block valves for the unit's two power operated relief valves (PORV's) were opened to allow PORV actuation for RCS pressure relief. However, leakage through the PORV seats caused a decrease in RCS pressure. The block valves failed to fully close when actuated from the control room, resulting in automatic safety injection and startup of emergency diesels due to low RCS pressure.

The NRC resident inspector was present in the control room, and the Region II Incident Response Center was manned. Additional regional support, including a supervisor, are enroute to the site for more detailed reviews.

The licensee has repaired the seal injection line gasket. Region II is reviewing safety concerns associated with the event. CP&L has agreed, and Region II has confirmed the agreement in writing, not to restart until safety questions have been resolved.

The licensee issued a news release. The NRC does not plan to issue a news release.

The State of South Carolina has been informed.

Region II (Atlanta) received notification of this occurrence by telephone from the resident inspector at 2:45 p.m. on November 30, 1981.

This information is current as of 2 p.m. on December 1, 1981.

Contact: C. A. Julian, RII 242-5538; C. W. Burger, RII 242-5538

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~~8112070301~~ PDR/LPDR

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

December 1, 1981

MEMORANDUM FOR: H. Denton
E. Case
D. Eisenhut
R. Purple
S. Hanauer
R. Vollmer
R. Mattson
J. Kramer
DL Assistant Directors
J. Sniezek
C. Michelson
D. Crutchfield

THRU: Steven A. Varga, Chief, ORB#1, DL SV

FROM: William J. Ross, Project Manager, ORB#1, DL

SUBJECT: DAILY HIGHLIGHT

H. B. Robinson Steam Electric Plant Unit No. 2

On Monday, November 30, 1981, a water leak occurred in the charging pump area as the result of ~50 gpm leak (body to bonnet) in a valve in a line to the reactor coolant pump seals. Approximately 1000 to 1500 gallons of water spilled into the charging pump area and overflowed to the floor of the auxiliary building. A local emergency was declared in these areas and partial evacuation of personnel initiated. The leak was isolated and the event terminated in four hours. Recovery from low (10%) pressurizer water level was aggravated by malfunctioning pressurizer block and relief valves. Safety injection was initiated but restoration of pressurizer level was achieved mainly through use of the charging pumps. During the transient three additional events occurred: one diesel did not function properly upon safety injection initiation; the bellows in the relief valve of the let-down line ruptured; and a telephoned threat of a bomb explosion at 5:00 p.m. was received. The plant had been in hot shutdown mode since November 6, 1981, and the licensee plans to continue preparation for startup as soon as failed components are repaired. IE plans to issue PNs for the transient and bomb threat.

William J. Ross

William J. Ross, Project Manager
Operating Reactors Branch #1
Division of Licensing

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