UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 30, 2000

MEMORANDUM TO: Charles A. Casto, Director
Division of Reactor Safety
Region II

FROM: Suzanne C. Black, Deputy Director Suzanne Black
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT 99-13, BRUNSWICK LICENSING BASIS FOR CONSIDERATION OF BYPASS LEAKAGE IN CONTROL ROOM AND OFFSITE DOSE CALCULATIONS (TAC NO. MA6662)

By memorandum dated September 21, 1999 (Attachment 1), your office requested technical assistance from the Office of Nuclear Reactor Regulation (NRR) regarding NRC contractor concerns (included in Attachment 1) about the Brunswick licensing basis associated with the calculation of control room and offsite radiation doses for a design basis loss-of-coolant accident (LOCA). The section of the Region II inspection report discussing this issue is also included in Attachment 1.

Specifically, your office asked the staff to address the following questions:

(1) Are source term assumptions for noble gases and iodine in Updated Final Safety Analysis Report (UFSAR) Section 15.6.4.3.1 consistent with the latest staff-approved Safety Evaluation (SE) for offsite doses and control room habitability? These values were 0.08 percent core iodine and 0.45 percent for noble gases available for release from containment. The contractor's report input stated that these values appeared to be significantly lower than those specified in Regulatory Guide 1.3.

(2) The contractor's report input stated that bypass leakage was not considered in the licensee's calculations for offsite and control room dose assumptions. The report input defined bypass leakage as main steam isolation valve (MSIV) leakage, primary containment inleakage, and secondary containment bypass leakage. Are the licensee's considerations of bypass leakage in offsite dose calculations consistent with the licensing basis for Brunswick?

(3) Are the results of the licensee's dose calculations in the UFSAR (listed below) realistic, or are they orders of magnitude lower than those reported by other boiling water reactors (BWRs)? The doses include:

Maximum control room doses during LOCA conditions listed in UFSAR Table 15.6.4-9.

DFX2

Two-hour site boundary doses shown in UFSAR Section 15.6.4.4 for design-basis analysis LOCA exposures.

(4) Did the modification which rerouted the discharge from the high-pressure coolant injection (HPCI) and reactor core isolation coolant (RCIC) drain pots from the plant stack to the condenser constitute a change to the plant licensing basis and a potential unreviewed safety question?

The responses to these questions follow.

Question 1

The latest staff-approved SE involving a LOCA was associated with the power uprate amendment. This amendment was issued November 1, 1996. This amendment was based upon licensee submittals beginning April 2, 1996, with numerous supplements, the last being October 29, 1996. It appears that the staff's assessment in the radiological dose area was based upon the General Electric (GE) Proprietary Information Report NEDC-32466P, "Power Uprate Analysis Report for Brunswick Steam Electric Plant Units 1 and 2." Table 9-3 of that report provides the initial inventory fractions within containment atmosphere. The Table indicates that 100 percent of the core noble gases and 25 percent of the iodides are in containment. These are not the values listed in Section 15.6.4.3.1.

It should be noted, however, that in the time frame of the late 1960's and early 1970's BWRs, preliminary safety analysis reports and FSARs included in their accident analyses two sets of source terms. One set was labeled "realistic" and the second was labeled "conservative." The section under question, Section 15.6.4.3.1, reflects the "realistic" source term. Another section, Section 15.6.4.5.2, makes reference to the TID-14844 source term, which is the "conservative" source term. This is consistent with the GE topical report.

Question 2

The licensee's considerations of bypass leakage in dose calculations are consistent with the licensing basis for Brunswick. Licensee's submittals dated December 30, 1980, August 30, 1985, and Table 9-3 of GE Report NEDC-32466P all indicate that MSIV leakage was not considered in the evaluations and that at no time did the licensee ever consider the occurrence of leakage from containment directly to the environment without filtration by the standby gas treatment system.

Question 3

The following Table provides a comparison between the Brunswick LOCA doses (those being questioned by your office) and those of other BWRs.

Table 3-1

	Exclusion Area Boundary		Control Room	
Plant	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Brunswick	7.8×10^{-6}	5.6×10^{-5}	2	0.003
Browns Ferry	16	0.3	21	1
LaSalle	32	3.6	20	1.6

With respect to the above Table, some comments are appropriate. The doses for Brunswick for the exclusion area boundary (EAB) and control room reflect different source terms. The control room reflects TID-14844 while the EAB reflects the "realistic" source term defined in Question 1. The Brunswick control room doses are at least an order of magnitude lower due to the exclusion of MSIV leakage from the calculations. Control room doses would also be affected by the quantity of unfiltered inleakage, control room ventilation system configuration and filtration characteristics, and site-specific atomospheric dispersion.

Question 4

The rerouting of the HPCI and RCIC drain pots from the reactor building equipment drain tank to the main condenser did result in a change to the plant's licensing basis and a potential unreviewed safety question.

Please contact Allen Hansen of my staff at (301) 415-1390 if you have any questions regarding this response.

Docket Nos. 50-324 and 50-325

Attachment: As stated

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* No substantive change to response memorandum ** See previous concurrence

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Question 4

The rerouting of the HPCI and RCIC drain pots from the reactor building equipment drain tank to the main condenser did result in a change to the plant's licensing basis and a potential unreviewed safety question.

Please contact Allen Hansen of my staff at (301) 415-1390 if you have any questions regarding this response.

Docket Nos. 50-324 and 50-325

Attachment: As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

September 21, 1999

MEMORANDUM TO: John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Bruce S. Mallett, Director
Division of Reactor Projects

SUBJECT: TASK INTERFACE AGREEMENT (TIA 99-13) BRUNSWICK LICENSING BASIS FOR CONSIDERATION OF BYPASS LEAKAGE IN CONTROL ROOM AND OFFSITE DOSE CALCULATIONS

During an inspection at the Brunswick facility, documented in NRC Inspection Report 50-325, 324/98-14, an NRC contractor inspector raised concerns regarding the licensing basis associated with calculation of control room and offsite radiation doses for a design basis LOCA. The inspector documented his concerns in the report input he provided to Region II. A copy of the report input is attached (Attachment 1). The licensee's position was that the plant meets their licensing basis, in part, based on NRC Safety Evaluations (SE) dated October 18, 1983 and February 16, 1989, to address control room habitability (NUREG-0737 ITEM III.D.3.4).

The issue regarding the control room and offsite dose calculations was identified by the contractor during review of a modification which rerouted the drain lines from the HPCI and RCIC turbine drain pots from the equipment drain tank to the main condenser. Since the equipment drain tank discharges directly to the plant stack without being processed by the standby gas treatment system, the effect of this modification was to potentially increase ground level release of radiation. This issue was identified as IFI 98-14-04 in Inspection Report (IR) 50-325, 324/98-14. The section of the IR which discussed the issues raised by the inspector is attached (Attachment 2).

The Region requests technical assistance in the evaluation of the contractor's concerns. Specific questions to be addressed to resolve the concerns are as follows:

1. Are source term assumptions for noble gases and iodine in UFSAR Section 15.6.4.3.1 consistent with the latest staff approved SE for offsite doses and control room habitability. These values were 0.08% core iodine and 0.45% for Noble gases available for release from containment. The contractor's report input stated that these values appeared to be significantly lower than those specified in RG 1.3.

2. The contractor's report input stated that bypass leakage was not considered in the licensee's calculations for offsite and control room dose assumptions. The report input defined bypass leakage as MSIV leakage, primary containment leakage, and secondary containment leakage. Are the licensee's considerations of bypass leakage in offsite dose calculations consistent with the licensing basis for Brunswick?

3. Are the results of the licensee's dose calculations in the UFSAR listed below, realistic, or are they orders of magnitude lower than those report by other BWRs? The doses include:

   - Two-hour site boundary doses shown in UFSAR Section 15-6.4.4 for DBA LOCA exposures.
   - Maximum control room doses during LOCA conditions listed in UFSAR Table 15.6.4-9.

4. Did the modification which rerouted the discharge from the HPCI and RCIC drain pots from the plant stack to the condenser constitute a change to the plant licensing basis and a potential unreviewed safety question?

This request was discussed between Herbert Berkow of the NRR staff and Ken Barr of Region II. If you have any questions contact Ken Barr (404) 562-4653.

Docket Nos. 50-325, 50-324
License Nos. DPR-71, DPR-62

Attachments: As stated (2)

cc w/atts:
A. Blough, RI
G. Grant, RIII
K. Brockman, RIV
H. Berkow, NRR
B. Bonser, DRP
G. West, DRP
J. Barnes, DRP

REPORT INPUT FROM CONTRACT INSPECTOR

HPCI/RCIC Drain Pot Drain Line Reroute to Main Condenser

In 1982, the drain lines from the HPCI and RCIC turbine drain pots were modified by rerouting them from the reactor building equipment drain tank to the main condenser. (The Unit 2 HPCI modification reviewed by the team was 82-138, dated February 15, 1984. The same modifications was also performed on Unit 1). The purpose of the change was to decrease the loading on radwaste systems to which the drain tank was pumped when it reached high level and to remove a source of high temperature in the tank area during normal operation that was caused by the discharge of high temperature condensate to the tank from the drain pots.

The team reviewed this modification and identified a potentially adverse impact that it could have on offsite and control room accident doses. This impact would be most significant for a large break LOCA, where these systems would have no active role, but where their passive function of containment isolation would be very important.

The design intent for the containment isolation function for these systems was that the HPCI and RCIC primary containment isolation valves would close for a large break LOCA. Any leakage past these valves would be contained in the systems themselves, and any system leakage from the systems into the secondary containment that became airborne would be processed by the standby gas treatment system (SGTS) before release to the environment through the plant stack. The SGTS was designed for 99% iodine removal efficiency.

This modification appeared to circumvent that design intent in that it created what appeared to be new release paths that bypassed secondary containment. Any leakage past the steam line containment isolation valves could potentially proceed unimpeded through this new path. Since the rerouted lines were non-seismically designed, non-safety-related components for which credit could not be taken for their integrity, such leakage would have to be considered as a direct, ground-level release to the environment. Not only did this path not have the radiation reduction benefit of dilution and holdup in the secondary containment and cleanup by the SGTS, it was also a direct pathway from the reactor core without the benefit of plateout and dilution inside the primary containment. Therefore, this modification appeared to have a significant potential to increase the consequences of an accident. The safety evaluation that was performed for this modification did not recognize or address this potential.

In order to understand the significance of this modification, the effect of the other major unfiltered leakage source, the main steam isolation valves (MSIVs), on the accident radiation exposures must be understood. In a typical BWR, the maximum allowable MSIV leakage rate was 11.5 scfh. Assuming single failure of one MSIV allowing 11.5 scfh leakage in that line and a total minimum pathway leakage on the remaining three lines producing typically another 11.5 scfh, the total unfiltered MSIV leakage would be 23 scfh. This normally accounts for about 95% of the total calculated exposures, which are usually a large fraction of the regulatory limits. At Brunswick, the total allowable unfiltered leakages through the HPCI and RCIC steam line containment isolation valves was 13 scfh, which should have been added to the MSIV leakages in order to correctly account for unfiltered leakage. This would have significantly increased the accident radiation exposures.

The licensee responded that this consideration was not required because the Brunswick licensing basis did not require accounting for leakage that bypassed the secondary containment. Discussion of this point led to discovery that *no* bypass leakage was considered in the licensee's analyses for offsite and control room accident doses, and as a result, their calculated doses reported in the UFSAR were orders of magnitude less than other comparable plants. This overall finding is discussed in more detail in a later report section.

10 CFR 50.59, "Changes, tests and experiments" required that licensees determine if changes to the facility as described in the SAR could increase the consequences of an accident and thereby involve an unreviewed safety question. Contrary to this requirement, the licensee performed a modification to the HPCI and RCIC systems' steam line drain pot drain lines to reroute them from the reactor building equipment drain tank to the main condenser and failed to consider in the 10 CFR 50.59 safety evaluation the potential that these changes could increase the offsite and control room radiation dose consequences for a design basis LOCA.

In reviewing this concern, it was discovered that the original drain line design, with routing to the equipment drain tank, also bypassed the secondary containment. The drain tank vent line penetrated the secondary containment and was piped directly to the radwaste building HVAC exhaust duct system. Although this arrangement also allowed unfiltered release directly from the core to the environment, which was not an acceptable design, the HVAC exhaust was through the plant stack, which did provide an elevated release, which would likely provide lower accident consequences than the ground level release that would result from the modified drain lines.

<u>Accident Radiation Dose Analyses Inconsistencies</u>

As noted in Section E1.1.b of this report, in reviewing design modification 82-138, which rerouted the HPCI and RCIC steam line drain pot drain lines from the reactor building equipment drain tank to the main condenser, the team discovered the that the licensee did not consider any bypass leakage in determining offsite and control room radiation exposures for design basis accidents. (Bypass leakage is defined as unfiltered primary containment leakage that bypasses the secondary containment and its cleanup system, SGTS.)

Typically, the highest impact bypass leakage paths in BWRs were those that connected directly to the reactor, such as the main steam isolation valves (MSIVs), and in the case of this plant, with the HPCI steam line drain pot drain lines rerouted to the main condenser, the HPCI and RCIC steam line isolation valves. Such paths allowed direct unfiltered, undiluted leakage from the reactor to the environment, and typically constituted approximately 95% of the total accident dose contributors. Not considering these paths would yield accident dose analyses that would be extremely non-conservative and unrealistic.

Additionally, the source term assumption used in the licensee's analyses were orders-of-magnitude less conservative that the requirements of Regulatory Guide 1.3, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" as follows:

Accident Dose Analyses Source Term Assumptions

	R.G. 1.3 Req'mt	Licensee Assumption
Core iodine available for	25%	0.08% (0.3% of release from containment required amount)
Noble gases available for	100%	0.45% (0.45% of release from containment required amount)

As a result of not considering major bypass leakage sources and using extremely non-conservative source terms, the accident exposures that were reported by this licensee in the UFSAR and previous FSARs were unrealistic and orders-of-magnitude lower than those reported by comparable plants. The following tables compare the licensee's UFSAR currently reported offsite and control DBA LOCA exposures, typical BWR values, and regulatory limits:

2-Hour Site Boundary Doses

	10 CFR 100 limits	Typical BWR	Brunswick
Thyroid	300 rem	250-290 rem	7.8×10^{-6} rem
Whole-Body	25 rem	10-20 rem	5.6×10^{-5} rem

Control Room Doses

	10 CFR 50, Appendix A,	Typical BWR	Brunswick	Criterion 19 limits
Thyroid	30 rem	20-25 rem	1.73 rem	
Whole-Body	5 rem	3-4 rem	0.416 rem	

The licensee maintained that the plant's licensing basis did not require consideration of bypass leakage based on their interpretation of Regulatory Guide 1.3., Paragraph C.1.e. of this document. This paragraph stated, "The primary containment should be assumed to leak at the leak rate incorporated or to be incorporated in the technical specifications for the duration of the accident. The leakage should be assumed to pass directly to the emergency exhaust system [SGTS] without mixing in the surrounding reactor building atmosphere and should be then assumed to be released as an elevated plume for those facilities with stacks." The licensee took the position that since the regulatory guide made no mention of how to treat bypass leakage it was not required to be considered, and that since these sentences in the regulatory guide required all primary containment leakage be assumed to pass through the SGTS, then consideration of bypass leakage was specifically not required since it could not pass through the SGTS due to the plant layout.

The team maintained that these sentences were meant to promulgate a requirement for the extra level of conservatism of not taking credit for holdup time or dilution in the reactor building in modeling one of the leakage pathways, the primary containment-to-secondary containment-to-SGTS pathway. This did not preclude the requirement that all other actual pathways be considered.

The licensee's interpretation was illogical in view of the actual plant configuration. It was also inconsistent with other licensing bases, the licensee's own procedures, NRC guidance, and industry precedent. For instance, Technical Specifications SR3.6.4.3.2 and 5.5.7 required periodic testing of the SGTS filter train to assure, among other things, that leakage around the filters was maintained below very low limits. Procedure 1(2)PT-15.1.2A, Rev 6, August 14, 1998, "Standby Gas Treatment Train 1A Filter Test", stated that its purpose was to verify that "...the system contains no bypassing that would compromise the function of the filter." Not allowing bypass leakage past SGTS filters was totally inconsistent with not considering unfiltered leakage from a much more concentrated source through the other bypass leakage pathways.

The NRC's concern with unfiltered bypass leakage was also demonstrated by NRC Information Notice 91-56, dated September 19, 1991, "Potential Radioactive Leakage to Tank Vented to Atmosphere". It was clear from in document that the NRC intended that all possible leakage pathways, including bypass leakage pathways, be accounted for as well as minimized. NUREG 0737, May 4, 1981, Section III.D.1.1, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors", required that licensees establish programs to minimize leakage of such systems because of its inordinate impact on accident doses, and Technical Specification 5.5.2 reflected this requirement. Not accounting for such leakage would be entirely inconsistent with these requirements.

Due to the late discovery of some elements of this concern and its vast scope, there was insufficient inspection time to effect its resolution. Therefore, this concern is identified as an unresolved item.

PORTION OF INSPECTION REPORT NUMBER 50-325,324/98-14 WHICH DISCUSSES TREATMENT OF BYPASS LEAKAGE. THE BASIS FOR THIS SECTION OF THE REPORT WAS THE INPUT RECEIVED FROM THE CONTRACTOR. AN IFI WAS IDENTIFIED PENDING A REVIEW BY NRC (NRR) OF THE CONCERNS EXPRESSED BY THE CONTRACTOR.

HPCI/RCIC Drain Pot Drain Line Reroute to Main Condenser

In 1982, the drain lines from the HPCI and RCIC turbine drain pots were modified by rerouting them from the reactor building equipment drain tank to the main condenser. The purpose of the change was to remove a source of high temperature water from the drain tank during normal operation resulting from the discharge of high temperature condensate to the drain tank from the drain pots.

During review of this modification, the team questioned whether this modification circumvented the design intent in that it created what appeared to be new release paths that bypassed secondary containment. Any leakage past the HPCI and RCIC steam line containment isolation valves could potentially proceed unimpeded through this new path. Therefore, this modification appeared to have a potential to increase the consequences of an accident. The safety evaluation that was performed for this modification did not recognize or address this potential.

The licensee responded that this consideration was not required because the Brunswick licensing basis did not require accounting for leakage that bypassed the secondary containment. Discussion of this point led to discovery that bypass leakage was not considered in the licensee's analyses for offsite and control room accident doses.

The potential that these changes could increase the offsite and control room radiation dose consequences for a design basis LOCA had apparently not been considered in design of the modification. The team concluded that additional review of the radiation control aspects of this modification was required. Pending completion of this review, this issue was identified to the licensee as Inspector Follow-up item 50-325,324/98-14-04, Consideration of Bypass Leakage in Control Room and Offsite Dose Calculations.

Attachment 2