Nebraska Public Power District
Nebraska's Energy Leader

NLS2000029
March 20, 2000

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

Subject: Design Basis Accident Radiological Assessment Calculational Methodology - Response to Request for Additional Information Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46

References:

1. Letter to Mr. J. H. Swailes (Nebraska Public Power District) from Lawrence J. Burkhart [signed by Robert A. Gramm] (U.S. Nuclear Regulatory Commission) dated March 6, 2000, Cooper Nuclear Station - Request for Additional Information (TAC No. MA7758).

2. Letter to U.S. Nuclear Regulatory Commission (NLS990122) from John H. Swailes (Nebraska Public Power District) dated December 22, 1999, Design Basis Accident Radiological Assessment Calculational Methodology Revision.

By letter dated March 6, 2000 (Reference 1), the Nuclear Regulatory Commission (NRC) requested the Nebraska Public Power District (District) to provide additional information on the Design Basis Accident Radiological Assessment Calculational Methodology Revision submitted by the District on December 22, 1999 (Reference 2). Attachment 1 provides the additional information requested.

Reference 2 included six calculations. Based on the information provided in Attachment 1, two of the calculations are not impacted [Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Meteorological Dispersion-Accident Analysis (Nuclear Engineering Design Calculation (NEDC) 99-036) and Dose Calculation for Control Room, EAB, and LPZ for a Main Steam Line Break (NEDC 99-035)]. It was agreed in discussions held with the NRC staff during the week of March 13, 2000, that three calculations will be revised and provided under separate letter by March 24, 2000 [Control Room, EAB and LPZ Doses Following a Control Rod Drop Accident (NEDC 99-034), Control Room, EAB and LPZ Doses Following a Loss of Coolant Accident (NEDC 99-033), and X/Q Values for Control Room Intake Using ARCON96 (NEDC 99-031)]. The status of the remaining calculation [Control Room Habitability and Offsite Dose for a Fuel Handling Accident (NEDC 99-032)] will be also be addressed in the March 24th letter.

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Reference 1, Question 6 requests justification for crediting iodine removal in the main turbine condenser. While the District believes that crediting iodine removal in the existing main turbine condenser design is already a part of the CNS licensing basis for radiological assessment calculation accident mitigation, the District also believes that it is appropriate to evaluate the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser and the main turbine condenser to confirm that these components will remain structurally intact (e.g., will not suffer gross structural failure) following a Safe Shutdown Earthquake.

The District will submit a letter, by March 24, 2000, describing the structural robustness of the existing main turbine condenser and the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser. The District will also address the low probability of needing the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser, and the main turbine condenser, for accident mitigation. In addition, this letter will provide a proposed license condition addressing when additional information will be provided to the NRC regarding the ability of the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser, and the main turbine condenser, to remain functional during and after a Safe Shutdown Earthquake.

Should you have any questions concerning this matter, please contact Sharon Mahler at (402) 825-5236.

Sincerely,

John H. Swailes
Vice President of Nuclear Energy

/rlb

Attachment

cc: Regional Administrator w/attachment
USNRC - Region IV

Senior Project Manager w/attachment
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachment
USNRC

NPG Distribution w/o attachment

Response to NRC Request for Additional Information Regarding Cooper Nuclear Station Design Basis Accident Radiological Assessment Calculational Methodology Submittal

The following is the Nebraska Public Power District's (District's) response to the Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI) dated March 6, 2000. The RAI questions are associated with the District's Design Basis Accident Radiological Assessment Calculational Methodology Revision submitted by the District to the NRC on December 22, 1999.

Question 1. Although the nuclides of interest in design bases accident analyses reach equilibrium values early in a fuel cycle, extended burnups can affect the core inventory. A substantial fraction of the energy produced during the final fuel irradiation cycle may be derived from Pu-239. The most significant difference in terms of radiological analyses is approximately 27 percent greater I-131 yield from Pu-239 fissions as compared with that for U-235 fissions. TID-14844 values were based upon a simplified formula that did not consider nuclide ingrowth. Because of these considerations **(a)** justify the AXIDENT source term for the extended burnup fuel design to which it is being applied. **(b)** Please insure that the AXIDENT source term is conservative with respect to the limiting design parameters of the fuels to be used (including the GE14 fuel) or modify the source term to be consistent with the most limiting fuel design. **(c)** If another source term is used in this analysis please provide details about how the source term was generated.

District response: *<"a, b, c" references added to the Question>*

a. The AXIDENT source term is based on TID-14844 values/methodologies, with extended fuel burnup corrections applied in accordance with NUREG/CR-5009.

b. The extended burnup fuel designs utilized at Cooper Nuclear Station (CNS) (including GE-14) are within the applicability of NUREG/CR-5009. No additional modification of the source term is necessary to accommodate the most limiting fuel design.

c. No other source term is proposed to be utilized.

Question 2. Page 16 of 108 of calculation NEDC 99-033, "Control Room, EAB, and LPZ Doses Following a LOCA," assumes single failure of the filter heater power. An operator action is credited to shut off the train with the failed filter power at 1 hour. Provide justification for the operator's action. This justification should identify the procedures that direct this operator's action, and the indication the operator uses to identify the failed train.

District Response:

Following an automatic initiation of the Standby Gas Treatment System (SGTS) on a Group VI isolation (Low Reactor Water Level or High Drywell Pressure for a LOCA), CNS Procedure 2.1.22 directs the Control Room Operator to align the SGTS per CNS Procedure 2.2.73. CNS Procedure 2.2.73 directs the Control Room Operator, in response to an automatic initiation of SGTS, to first ensure that the Reactor Building to atmosphere DP is maintained at less than or equal to -0.25" wg, and to then place the preferred SGTS fan in operation and the other SGTS fan in a standby condition. The Control Room Operator has the following indications and alarms available at the SGTS Control Room Panel, which would indicate a filter heater power failure:

-- SGT HIGH MOISTURE audible and visual alarm, which indicates the relative humidity of the airstream entering the SGTS charcoal filter exceeds 70%.

-- Moisture indicator, which provides indication of the airstream moisture content entering the adsorber element.

-- Moisture indicator which provides indication of the moisture content of the airstream leaving the heaters.

Upon receipt of the SGT HIGH MOISTURE ALARM, the Control Operator is directed by CNS Procedure 2.3.2.15 to check operation of the SGTS heater and moisture separator, align to the other SGTS train, and secure the SGTS train causing the alarm.

Following receipt of the NRC Safety Evaluation Report (SER) for the December 22, 1999 request, the District will revise procedure(s), as required, to specify the required 1 hour alignment time.

Question 3. Please clarify whether no mixing or partial mixing in the secondary containment is modeled in the radiological dose analyses.

District Response:

Where applicable, the dose consequence analyses specify which assumption is used. The Fuel Handling Accident (FHA) calculation credits mixing in the refueling area volume. The Loss of Coolant Accident (LOCA) calculation was performed twice to compare a secondary containment mixing scenario (Standard Review Plan 6.5.3 case) with an instantaneous release case (no secondary containment mixing as discussed in Regulatory Guide 1.3). The LOCA calculation involving the instantaneous release scenario is the calculation being proposed.

Question 4. The Cooper Ventilation Filter Testing Program that tests ESF filters allows a system bypass of less than 1% for in place tests. The Cooper radiological analyses do not consider this bypass. This seems non-conservative. Please justify this apparent non-conservatism or model it in your analyses.

District Response:

The District will revise the radiological analyses to include an additional 1% bypass flow.

Question 5. On page 13 of 28 in calculation NEDC 99-034, "Control Room, EAB, and LPZ Doses Following a CRDA," please clarify assumption 6.16. This assumption states that the Control Room isolation is ignored in the analysis. Other parameters in the analysis infer isolation of the Control Room. In your clarification please state at what point the isolation is credited.

District Response:

Assumption 6.16 will be revised to clarify that, for the purpose of maximizing the Control Room operator dose calculation, the Control Room ventilation system is assumed to shift to the Control Room Emergency Filtration Mode 24 hours after the Control Rod Drop Accident (CRDA) is assumed to occur. This traps the radioactivity present in the Control Room at time equal to 24 hours, thus analytically maximizing dose to the Control Room operators.

Question 6. In the radiological analyses used to support the proposed change credit is taken for plateout on the condenser. Per page 8 of 108 of NEDC 99-033, it is inferred that the condenser credited is a non-seismic condenser. The staff is not aware of situations where non-seismic steam line piping and condensers are credited for iodine removal. After an extensive review, the staff has previously approved a methodology that credits iodine removal after a Safe Shutdown Earthquake. This methodology is given in "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," NEDC-31858P-A. Please justify why this standard methodology is not used for crediting iodine removal for your application. Also, please justify the methodology used.

District Response:

To be provided under separate response letter.

Question 7. Cooper used NEDO-31400 to remove the main steam line radiation monitor (MSLRM) scram function and main steam isolation valve (MSIV) isolation function. As part of that review the impact of bypassing the offgas treatment system until late in the power ascension should have been addressed per NEDO-31400. The source term for Cooper at that time was smaller than the proposed TID source term. If Cooper's operating procedures continue to allow bypassing of the offgas treatment system until late in the power ascension the impact of the new source term needs to be addressed. Are the offgas pretreatment and post-treatment radiation monitors currently utilized to isolate the offgas treatment line and/or the offgas process line before the acceptable release rates are exceeded? If this condition applies at Cooper, then these monitors should automatically isolate the process line. Please note that according to NEDO-31400, plants that do not have the capability to bypass the treatment system do not have the additional requirement of automatic isolation of the process line.

District Response:

The CRDA source term will not reach the environment via the offgas treatment system flow path.

At low reactor power levels, when Steam Jet Air Ejectors (SJAEs) are not in service and the mechanical vacuum pumps are used to remove noncondensables from the condenser, a Main Steam Line high radiation signal resulting from the CRDA would trip the mechanical vacuum pumps and close the mechanical vacuum pump inlet and outlet valves. Thus, there would be no mechanical motive force to draw the noncondensables from the condenser into the offgas system at low power levels.

When the SJAEs are in service a 30 minute hold-up line downstream of the SJAE exhaust provides for decay of fission gases. The 30 minute hold-up line does not have a bypass. A CRDA would cause a SJAE off-gas radiation monitor high radiation signal to initiate a 15 minute timer which, after 15 minutes, will isolate the off-gas system downstream of the 30 minute hold-up line. Thus the fission gas released from the CRDA will be isolated prior to exiting the 30 minute hold-up line.

Question 8. Page 12 of 108 of calculation NEDC 99-033, "Control Room, EAB, and LPZ Doses Following a LOCA," identifies an extrapolation factor formula for laminar flow taken from ORNL-NSIC-5 (page 10-52). The staff is not aware that this formula has been previously utilized by the staff to determine MSIV leakage. If this methodology has not been previously accepted by the staff, further justification will be needed to evaluate its acceptability. Please provide any information regarding past utilization and acceptance of this methodology by the staff.

District Response:

A simplified adjustment based on a "PV=nRT" relationship will be used in place of the ORNL methodology.

Question 9. The radiological analysis takes credit for alternate source term insights (NUREG-1465). Unless the amendment is filed under 10 CFR 50.67, the insights of TID-14844 should be used (or your current licensing basis). If TID-14844 is used, the timing aspects of the source term being instantaneously available should be considered in the modeling of the accident. For example, any delayed actuation of the standby gas treatment system or Control Room HVAC [heating, ventilation, and air conditioning] should be considered. Please indicate which source term methodology will be used and make the analyses consistent with this choice.

District Response:

The NUREG-1465 delayed fuel failure assumption previously used in the LOCA calculation will be deleted and replaced with the assumption that the TID-14844 source term is immediately available for release to primary containment.

For the primary to secondary containment leakage path, the TID-14844 source term will be assumed to pass directly into the standby gas treatment system without mixing in the surrounding reactor building atmosphere, and then assumed to be released as an elevated plume. These revised assumptions are consistent with Regulatory Guide 1.3 assumptions and the LOCA analysis described in the CNS Operating License NRC SER (Section 15.2) for Cooper Nuclear Station.

For primary containment leakage through the MSIVs the TID-14844 source term will be assumed to be immediately available for leakage through the MSIVs. The MSIV leakage will be released via the turbine-condenser complex.

Following receipt of the NRC SER for the December 22, 1999 request, the District will revise applicable procedure(s) to specify the manual initiation of the Control Room Emergency Filtration System within 20 minutes of the LOCA, as assumed in the LOCA calculation.

Question 10. On page 7 of 8 in calculation NEDC 99-032, "Control Room Habitability and Offsite Dose for a Fuel Handling Accident," it is stated that Scientech's calculated Control Room doses are increased 5% to conservatively bound modeling inaccuracies or future uncertainties. How was the 5% value determined?

District Response:

As discussed in NEDC 99-032, the mathematical model used to calculate the effective X/Q involves polynomial best fit curves of data. Polynomial best fit curves introduce small errors when evaluating over the entire range of the curve. It was noted that the best fit curve for the 60,000 cfm flow curve had one point on the curve which was about 2% higher than the corresponding point on the 50,000 cfm flow best fit curve. Although this could have been explained using error analysis associated with doing best fit polynomial curves, it is simpler to add 5% to the Fuel Handling Accident (FHA) dose calculation results to account for the minor polynomial best fit curve errors statistically introduced by the best fit methodology.

Question 11. NEDC 99-032, "Control Room Habitability and Offsite Dose for a Fuel Handling Accident," takes credit for 72 hours of decay. The current basis for Technical Specification 3.9.6 states that the decay time is a minimum 24 hours. By what means is the decay time controlled to be 72 hours or greater?

District Response:

The Technical Specification 3.9.6 Bases (referenced in the question) will be revised to reflect the assumption of the 72 hour decay time. In addition, the 72 hour decay time will be incorporated into the Updated Safety Analysis Report (USAR) and applicable station refueling procedures.

Question 12. On page 6 of 161 in calculation NEDC 99-032, "Control Room Habitability and Offsite Dose for a Fuel Handling Accident," the radionuclide release is 0.76 times the release from a Fuel Handling Accident (FHA) in a core of all 7x7 fuel bundles. Does this at a minimum equate to the release from the failure of all the fuel rods in one GE14 fuel bundle?

District Response:

The radioactive release in CNS calculation NEDC 99-032 does equate to the release from the failure of all the fuel rods in more than one GE14 fuel bundle. A GE14 fuel bundle contains 78 full length fuel rods and 14 partial length rods for an effective total of 87.33 full length fuel rods per bundle. The radioactive release in CNS calculation NEDC 99-032 equates to approximately 160 damaged GE14 fuel rods.

Question 13. In NEDC 99-032, "Control Room Habitability and Offsite Dose for a Fuel Handling Accident," "effective" X/Qs are determined for the Control Room. How does the final dose result compare to using Control Room X/Qs based on the site meteorological data?

District Response.

The site meteorological data is used in determination of the "effective" X/Q.

An "effective" X/Q was used to account for the change in dispersion factors and release rates as a function of time. In lieu of introducing a large number of time steps with varying release rates and dispersion factors, the analysis determines an equivalent X/Q, based on integration, that when used with a constant release rate produces an equivalent amount of activity entering the Control Room envelope. This approach is more mathematically correct as compared with inputting a step function. As a note, the equivalent dispersion factors used are based on the site meteorological data.

Dispersion factors (X/Q) as a function of Reactor Building exhaust flow using site meteorological data and a fan coast down curve were developed. Several curves plotting the product of the release rate and X/Q as a function of time were developed and integrated to determine the highest integrated concentration at the Control Room intake for each case considered. The "effective" X/Q was then determined such that the product of the maximum release rate, the duration of the release, and the "effective" X/Q equals the highest integrated concentration for each case.

Question 14. On page 12 of 37 in calculation NEDC 99-035, "Dose Calculation for Control Room, EAB, and LPZ for a MSLB," the flashed fraction of the liquid coolant is determined by assuming a constant enthalpy process. Why was the starting point for the calculation assumed to be 1000 psia instead of the normal operating pressure of 1054.7 psia?

District Response:

USAR Section XIV 6.5.6 (Reference 1 in the Main Steam Line Break (MSLB) calculation) derives the total amount of liquid which is discharged through the break prior to MSIV isolation. This section also states:

> "The steam flow-steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 60 psi/sec. The formation of bubbles in the reactor vessel water causes a rapid rise in the water level. The analytical model used to calculate level rise predicts a rate of about 6 feet/second. Thus, the water level reaches the vessel steam nozzles at 2 seconds after the break as shown in Figures XIV-6-13a and b. From that time on a two-phase mixture is discharged from the break."

Utilizing the above depressurization rate of 60 psi/sec, the 2-second time to reach two-phase flow, and the initial reactor operating pressure range of 1050 – 1100 psia, the two-phase flow mixture will initially be at a reactor pressure range of approximately 930 – 980 psia. Therefore, 1000 psia was used as a conservative value for the liquid flash fraction calculation in order to maximize the amount of energy available to flash the liquid phase into steam.

Question 15. Why was the Control Room dose due to the main steam line break accident not determined for a reactor coolant specific activity spike of 4.0 μCi/gm Dose Equivalent I-131?

District Response:

The CNS Licensing Basis for determining that Control Room dose is maintained within the limits of General Design Criterion (GDC) 19 of 10 CFR 50, Appendix A, is based on the Technical Specification Limiting Condition for Operation (LCO) value for continuous operation, which is 0.2 μCi/gm Dose Equivalent I-131. As such, a calculation to determine the Control Room operator dose using the reactor coolant activity spike of 4.0 μCi/gm Dose Equivalent I-131 is not required by the CNS Licensing Basis.

The most recent revision to the CNS license basis, involving the Dose Equivalent I-131 values, was associated with the conversion of the CNS Technical Specifications to Improved Technical Specifications (CNS Technical Specification Amendment #178, TAC NO. M98317). As a result of this Amendment, the Bases for LCO 3.4.6, "Reactor Coolant System Specific Activity," states:

> "The limits on the specific activity of the primary coolant also ensure the thyroid dose to the Control Room operators resulting from a main steam line break outside containment during steady state operations will not exceed the limits specified in GDC19 of 10CFR50, Appendix A."

This limit provided in LCO 3.4.6 requires that specific iodine activity is limited to less than or equal to 0.2 μCi/gm Dose Equivalent I-131.

In contrast, the evaluations associated with offsite doses and compliance with 10 CFR 100 are performed with both the continuous steady state value for reactor coolant specific activity (0.2 μCi/gm Dose Equivalent I-131), as well as with a spike of 4.0 μCi/gm Dose Equivalent I-131.

– Per Standard Review Plan 15.6.4 (Section II.1), Rev 2, for a MSLB with an assumed pre-accident spike iodine concentration corresponding to the maximum value allowed in the Technical Specifications (4.0 μCi/gm Dose Equivalent I-131 in CNS LCO 3.4.6, Action A) the calculated doses should not exceed the guideline values of 10 CFR 100.

-- Per Standard Review Plan 15.6.4 (Section II.2), Rev 2, for a MSLB with an assumed iodine concentration corresponding to the equilibrium value for the continued full power operation the doses should not exceed a small fraction (i.e. 10%) of 10 CFR 100 dose limits.

Consistent with the above, the Technical Specification Bases for LCO 3.4.6 state that the limits on the maximum allowable level of radioactivity in the reactor coolant (i.e., 4.0 μCi/gm Dose Equivalent I-131 in CNS LCO 3.4.6, Action A) are established to ensure that in the event of a release of any radioactive material to the environment during a Design Basis Accident (DBA), radiation doses are maintained within the limits of 10CFR100. Additionally, the Bases state that the limits on specific activity of the primary coolant during steady state operation ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from a MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10CFR100.

Question 16. Provide an overall evaluation of the quality of the meteorological data used in your December 22, 1999, submittal. Did the overall meteorological program meet the guidelines of Regulatory Guide 1.23, "Onsite Meteorological Programs"? If there were deviations, describe why the data were still deemed to be adequate to use in the analyses. The intent of this question is to assess the overall quality of the data. A detailed review of each data point is not expected.

District Response:

As described in the District's response to the NRC regarding NUREG-0737 Supplement 1 (letter dated November 15, 1984), the present CNS meteorological program was installed to comply with the recommendations of Regulatory Guide 1.23. The District noted in its response that the instrument accuracy limits for very large differential temperatures (greater than 5.28 degrees C per 100 meters) may be outside those recommended in Regulatory Guide 1.23. The District concluded that values of this magnitude are well beyond those expressed in Table 1 of the Regulatory Guide and would have no effect on the determination of atmospheric stability under these conditions. No other exceptions to the Regulatory Guide 1.23 were noted.

More recently, CNS contracted a vendor to conduct a meteorological inspection in April 1999. The following excerpts are taken from the vendor report regarding the differential temperature instrument accuracy:

-- [Vendor] compared stability class distributions for the previous 14 years of CNS meteorological data to determine if any differences in the data could be contributed to calibration of the data or were likely year to year climatological fluctuations.

-- The 100-meter wind and associated delta-t distributions for the 14 data years [1984-1998] were very consistent with only minor year-to-year differences. The largest differences occurred in 1995 where a large occurrence of A (unstable) stability was offset by a corresponding decrease in D (neutral) stability. A similar pattern to a lesser degree occurred in 1996 and is common in warm and dry years. Based on this review, there is no indication from the stability distributions that the delta-t data were biased by the calibration procedures for the 100- or 10-meter temperature.

-- As with the upper level wind and delta-t, the 10-meter wind and associated delta-t distributions for the 14 data years were very consistent with only minor year-to-year differences.

-- [Vendor] performed a thorough review of the CNS meteorological data, stability class distributions, and USAR data to determine if the CNS temperature calibration practices have had an affect on the meteorological data quality. Based on this review, there appears to be no effect on the long term meteorological data quality. Further, the delta-t data and associated stabilities are representative of the CNS site based on a comparison with the USAR data and the additional 14-year data record.

Meteorological instruments are calibrated and maintained per approved procedures. Meteorological data is reviewed, validated, and summarized for analysis each year to support submittal of the CNS Annual Operating Report- Radioactive Effluents to the NRC. Self assessments and/or Quality Assurance Program audits are periodically conducted to ensure the meteorological program is conducted per procedures and regulatory commitments. Such assessments and/or audits provide reasonable assurance that the quality of the meteorological data adequately meets the guidelines of Regulatory Guide 1.23 and regulatory commitments. Lastly, NRC inspections are conducted to ensure meteorological program compliance with NRC rules and regulations, and with the conditions of the CNS Operating License. During the most recent inspection conducted in September 1999 no findings were noted in the meteorological program. The inspection included reviews of calibration procedures and calibration records for meteorological monitoring instrumentation and meteorological instrument operability, reliability, and annual meteorological data recovery.

Question 17. Was delta-T data recovery during 1995 and 1996 below 90%? Throughout the 5-year period, were there occurrences of very unstable conditions, as defined by the delta-T measurements, during night time hours? If so, to what is this attributed?

District Response:

As reported in the Cooper Nuclear Station Annual Operating Report – Radioactive Effluents (letters dated April 29, 1996 and April 19, 1997), the delta-T data recovery for 1995 and 1996 is given below:

	1995	1996
100m - 10m Delta T	83.4%	84.8%
100m - 60m Delta T	95.0%	95.2%
60m - 10m Delta T	84.2%	84.2%

There were occurrences of very unstable conditions, as defined by delta-T, recorded occasionally during nighttime hours. These occurrences were attributed to weather conditions such as wind shifts and minor temperature fluctuations.

CNS contracted a vendor to conduct a meteorological assessment in April 1999. The following excerpts are taken from the vendor report:

-- [Vendor] has reviewed the stability data contained in Section 3.0 of the Updated Safety Analysis Report (USAR). The stability data are presented for two twelve-month periods; March 1970 through February 1971, and March 1971 through February 1972. Stability classifications are based on the 100- meter wind speed and the direction and are divided into four classifications, which include a combination of Pasquill-Gifford standard A-G classes; Unstable (A, B, and C combined), Neutral (D), Moderately Unstable (E), and Very Stable (F and G combined).

-- Joint frequency distributions of wind speed and direction versus stability class were compared between the USAR data and the 1998-1999 data period. Primary and secondary wind peaks of wind direction by stability class for the 1998-1999 period matched up well with the USAR data.

-- [Vendor] compared stability class distributions for the previous 14 years of CNS meteorological data to determine if any differences in the data could be contributed to calibration of the data or were likely year to year climatological fluctuations.

-- The 100-meter wind and associated delta-t distributions for the 14 data years [1984-1998] were very consistent with only minor year-to-year differences. The largest differences occurred in 1995 where a large occurrence of A (unstable) stability was offset by a corresponding decrease in D (neutral) stability. A similar pattern to a lesser degree occurred in 1996 and is common in warm and dry years. Based on this review, there is no indication from the stability distributions that the delta-t data were biased by the calibration procedures for the 100- or 10-meter temperature.

-- As with the upper level wind and delta-t, the 10-meter wind and associated delta-t distributions for the 14 data years were very consistent with only minor year-to-year differences.

-- A final comparison was made of the hourly stability classes and the joint frequency distributions using the 1997 and 1998 CNS meteorological data. The hourly stability classes were divided into three groups; day, night, and transition. The transition period is a the period one hour before and after sunrise/sunset where delta-t's and corresponding stabilities are changing rapidly. Although seasonally dependent, the typical hourly pattern expected would be stable (F and G) near sunrise moving rapidly to unstable (B and A) by late morning. The afternoon is typically unstable but begins to move toward neutral (D) near sunset, and finally back to stable (F and G) by midnight. Rainy, cloudy and windy days are characterized by neutral (D) to somewhat stable (E) conditions throughout the day or night. With some minor exceptions, this pattern was observed in the 1997 and 1998 data. Occasional occurrences of unstable conditions at night were evident in the summer months of both years but were associated with wind shifts and minor temperature fluctuations common during short summer nights.

-- [Vendor] performed a thorough review of the CNS meteorological data, stability class distributions, and USAR data to determine if the CNS temperature calibration practices have had an affect on the meteorological data quality. Based on this review, there appears to be no effect on the long term meteorological data quality. Further, the delta-t data and associated stabilities are representative of the CNS site based on a comparison with the USAR data and the additional 14-year data record.

Question 18. With respect to control room X/Qs, what is the basis for assuming a diffuse release from the turbine building? From where would releases be most likely to occur (vents, doors, and other potential openings to the environment)?

District Response:

During Turbine Building Ventilation System operation, the Turbine Building exhaust is directed to a common exhaust plenum (located east of the Turbine Building) by the Turbine Building exhaust fans (which would be de-energized on a loss of offsite power). The discharge of the plenum is approximately elevation 938 feet and is approximately 290 feet (88 meters) horizontally from the Control Room intake.

Following a postulated accident resulting in a radioactive release to the Turbine Building, in which there is no loss of offsite power (no LOOP), the Turbine Building exhaust fans continue to run, and the Turbine Building exhaust flow is directed vertically upward. In order for the release to reach the Control Room intake, the flow would have to rise above the turbine building roof (approximately elevation 1007 feet) and come down to the Control Room intake (approximately elevation 957 feet). Due to the vertical velocity, elevations, intervening building, and large horizontal distance to the Control Room intake, it is judged that this case would not yield bounding results for Control Room dose when compared to a postulated accident resulting in a radioactive release concurrent with a LOOP.

For the LOOP case, the Turbine Building exhaust fans would coast down and come to a stop, leaving no forced mode of ventilation to direct the Turbine Building atmosphere to the Turbine Building Ventilation system exhaust point. Leakage to the environment could be from any number of possible locations including the opening and closing of various doors, openings around door seals, duct penetrations, conduit penetrations, piping penetrations, and the turbine building siding itself. Leakage would therefore be expected to come from many locations along the perimeter of the turbine building external walls, not from any single location, as in the case of forced ventilation directed to a specific point. All faces have many potential leakage paths and each wall could be likely candidates for leakage.

Calculation NEDC 99-031 developed Control Room X/Q for turbine building release by conservatively selecting the wall closest to the Control Room intake as the leaky wall.

The calculation uses the entire surface area of this wall as the area source, in accordance with Section 2.5.7 of NUREG/CR-6331, Rev. 1. The NUREG gives the example for using the Diffuse Source Release model for the case where there are releases from many openings on a face of a building. Since this model was felt to most accurately describe the as-built layout of the plant and leakage openings, it was the one used in the calculation to model turbine building releases.

Based on followup discussions with the NRC Staff during the week of March 13, 2000, the initial horizontal and vertical diffusion coefficients used in determining the Turbine Building diffuse release X/Qs will be recalculated by dividing the assumed release area width, and the assumed release area height, by 6.

Question 19. With respect to the main steam line break assessment, **(a)** what is the basis for assuming a puff release to the environment? **(b)** Provide further details in the comparison between assuming a uniform and Gaussian distribution within the puff resulting in essentially the same integrated X/Q. **(c)** What assurance is there that the effluents will all pass relatively quickly as a puff?

District Response: *<"a, b, c" references added to the Question>*

(a) The main steam line break scenario is simulated as a puff release because it is a short-duration event for which the assumptions and theory of a continuous plume are not valid; i.e., the duration of the release for the main steam line break (~10 seconds) is shorter than the transport time between the source and receptor (~37 seconds). Standard Review Plan Section 15.6.4, paragraph II.3 and Standard Review Plan Section 2.3.4, paragraph III.1. provide additional details. Specifically, Standard Review Plan 2.3.4, paragraph III.1 states "Most accidental releases can be considered as continuous releases (i.e., on the order of several minutes or more). However, some releases such as from steam line breaks or of hazardous chemicals may be considered as instantaneous (puffs)." Additionally Section XIV, paragraph 6.5.8.3, of the Cooper Nuclear Station Updated Safety Analysis Report states "Since all of the activity is released to the environs in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated."

(b) Conservatively assuming that the entire release plume passes the Control Room intake prior to isolation of the intake (isolation occurs in 60 seconds), the integral of X/Q with respect to time (i.e., the dose) would be the same for the Gaussian distribution as for the uniform distribution.

(c) The assurance of relatively quick passage of the puff can be obtained from the comparatively small separation distance between the point of origin and the receptor (37 meters) and the wind climatology at that location. In the analysis presented (NEDC 99-035), 1 m/sec wind speeds were used as conservative low wind speeds (approximately 97% of the time, wind speeds exceed this value). This results in a time of 37 seconds for the leading edge of the plume to reach the Control Room intake (this ignores the initial velocity of the release). However, the calculation does not consider the 37 second delay. Instead, the calculation assumed a hemispherical cloud with transit time of 57.3 seconds for the release to pass over the intake. This fully exposes the cloud to the Control Room ventilation intake before it isolates in 60 seconds. The higher wind speeds that dominate at CNS would reduce the cloud exposure to the Control Room intake time considerably. Additionally, for the short travel paths involved, stagnation or recirculation of the wind is not likely.

Question 20. Are design flow rates and isolation times based on Technical Specification values?

District Response:

Few of the input values are found in Technical Specifications. For those values found in Technical Specifications, the input values used in the radiological analyses are the same as or, when determined to be more appropriate, more conservative than the Technical Specification value. When Technical Specification values were not applicable, design values, or more conservative values are used as appropriate. No flow rates or isolation times are used which are outside of the Technical Specification allowable values.

Question 21. For the fuel handling accident, what assurance is there that the air around the control room intake will be essentially free from effluents as soon as the release is routed to the standby gas treatment system (SGTS), that is, that the X/Q will instantaneously drop to 10E-9 sec/cu m?

District Response:

The instantaneous drop in X/Q is a modeling approximation that results from a new transport and diffusion time step. This step is associated with the shutdown of the normal reactor building ventilation (and its associated X/Q), and startup of the SGTS and its associated elevated release point X/Q. This is consistent with the modeling approximation that also assumes the Control Room intake will see the higher reactor building effluent release the instant the radioactivity leaves the reactor building vent.

Question 22. In several calculations the wind speed is assumed to be 1m/s. Approximately what percent of the time does this occur?

District Response:

Examinations of meteorological data show that the wind speed remains below 1 m/s for approximately 3% of the time.

Additional Verbal Question Provided After Receipt of RAI. The issue of not increasing suppression pool scrubbing effects is still applicable. Your calculations should not increase the factor (i.e., your current licensing basis factor of 2 should be maintained).

District Response:

Calculation NEDC 99-033 will be revised to ensure that suppression pool scrubbing does not exceed a factor of 2, which is consistent with the current licensing basis.

Correspondence No: NLS2000029

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COMMITMENT	COMMITTED DATE OR OUTAGE
Following receipt of the NRC SER for the December 22, 1999 request, the District will revise procedure(s), as required, to specify the 1 hour alignment time.	Prior to Startup from RFO19
The District will revise the radiological analyses to include an additional 1% bypass flow.	March 24, 2000
Assumption 6.16 will be revised to clarify that, for the purposes of maximizing the control room operator dose calculation, the control room ventilation system is assumed to shift to the Control Room Emergency Filtration Mode 24 hours after the CRDA is assumed to occur.	March 24, 2000
A simplified adjustment based on a PV=nRT relationship will be used in place of the ORNL methodology.	March 24, 2000
The NUREG-1465 delayed fuel failure assumption previously used in the Loss of Coolant Accident (LOCA) calculation will be deleted and replaced with the assumption that the TID-14844 source term is immediately available for release to primary containment.	March 24, 2000
The Technical Specification 3.9.6 Bases (referenced in the question) will be revised to reflect the assumption of the 72 hour decay time. In addition, the 72 hour decay time will be incorporated into the USAR and applicable station refueling procedures.	TS Bases Changes and Refueling Procedures - Prior to Startup from RFO19 USAR Changes - 120 days from receipt of SER
Calculation NEDC 99-033 will be revised to ensure that suppression pool scrubbing does not exceed a factor of 2, which is consistent with the current licensing basis.	March 24, 2000
Three calculations will be revised and provided under separate letter by March 24, 2000 [Control Room, EAB and LPZ Doses Following a Control Rod Drop Accident (NEDC 99-034), Control Room, EAB and LPZ Doses Following a Loss of Coolant Accident (NEDC 99-033), and X/Q Values for Control Room Intake Using ARCON96 (NEDC 99-031)].	March 24, 2000
Following receipt of the NRC SER for the December 22,1999 request, the District will revise applicable procedure(s) to specify the manual initiation of the Control Room Emergency Filtration System within 20 minutes of the LOCA, as assumed in the LOCA calculation.	Prior to Startup from RFO19

Correspondence No: NLS2000029 Page 2 of 2

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COMMITMENT	COMMITTED DATE OR OUTAGE
The status of the remaining calculation [Control Room Habitability and Offsite Dose for a Fuel Handling Accident (NEDC 99-032)] will be also be addressed in the March 24th letter.	March 24, 2000
The District will submit a letter, by March 24, 2000, describing the structural robustness of the existing main turbine condenser and the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser. The District will also address the low probability of needing the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser, and the main turbine condenser, for accident mitigation. In addition, this letter will provide a proposed license condition addressing when additional information will be provided to the NRC regarding the ability of the main steam line piping from the Main Steam Isolation Valves to the main turbine condenser, and the main turbine condenser, to remain functional during and after a Safe Shutdown Earthquake.	March 24, 2000
For the LOCA calculation the primary to secondary containment leakage path, the TID-14844 source term will be assumed to pass directly into the standby gas treatment system without mixing in the surrounding reactor building atmosphere, and then assumed to be released as an elevated plume.	March 24, 2000
Based on followup discussions with the NRC Staff during the week of March 13, 2000, the initial horizontal and vertical diffusion coefficients used in determining the Turbine Building diffuse release X/Qs will be recalculated by dividing the assumed release area width, and the assumed release area height, by 6.	March 24, 2000