NMC
Committed to Nuclear Excellence

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Kewaunee / Point Beach Nuclear
Operated by Nuclear Management Company, LLC

NRC-02-078

September 13, 2002

10 CFR 50.67

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

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Response to Request for Additional Information Related to Proposed Revision to the Kewaunee Nuclear Power Plant Design-Basis Radiological Analysis Accident Source Term

References:
1) Letter from Mark E. Warner (NMC) to Document Control Deck (NRC), "Revision to the Design Basis Radiological Analysis Accident Source Term," dated March 19, 2002.

2) Letter from John G. Lamb (NRC) to Mark E. Warner (NMC), "Kewaunee Nuclear Power Plant – Request for Additional Information Related to Proposed Revision to the Kewaunee Nuclear Power Plant Design-Basis Radiological Analysis Accident Source Term (TAC NO. MB4596), dated July 3, 2002.

In reference 2, the Nuclear Regulatory Commission (NRC) staff requested additional information concerning Nuclear Management Company's, LLC, (NMC's) submittal on use of alternative source term (AST) at Kewaunee nuclear power plant (Reference 1). This letter is NMC's response to the NRC's request for additional information.

Attachment 1 to this letter contains the questions the NRC staff requested with NMC's responses. Attachment 2 contains a figure showing a general plant arrangement as requested by the NRC staff.

During a telephone conversation with the NRC staff, a request was made to state the methodology to be used for KNPP's environmental qualification program. In reference 1, NMC stated that the alternate source term methodology would be implemented selectively. KNPP will apply AST methodology to design basis accidents to calculate offsite dose and control room dose. All other dose calculations, including equipment qualifications, will use the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," methodology.

A001

In reference letter 2, a 45-day response request was mutually agreed to. Per a telephone conversation with John Lamb, of your staff, an additional time was approved for this response.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on September 13, 2002.

Sincerely,

Thomas Coutu

Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant

GOR

Attach.

cc - US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

ATTACHMENT 1

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

September 13, 2002

NMC Responses to NRC Questions

NRC Question #1

Provide the radiological dose calculations performed for determining the radiological doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the control room for all design-basis accidents evaluated. If computer code programs were used for the dose calculations, provide copies of the inputs prepared and outputs obtained from the computer code system. If spreadsheets were used, provide copies of its calculation sheets.

NMC Response

In order to support the timely review of the Kewaunee submittal, Westinghouse has transmitted the calculations to its Rockville office where they are available for viewing. These calculations include input descriptions and copies of the important output from the computer codes.

NRC Question #2

In a letter to NRC dated February 28, 1989 (NRC 89-23), you stated that you performed an extensive system performance testing on the control room ventilation system to quantify unfiltered air inleakage to the control room. You further stated that the test estimated approximately 200 cfm of unfiltered inleakage into the control room emergency zone through identifiable pathways. Since 1989, the staff has been working toward resolution of generic issues related to control room habitability, with a particular focus on the validity of the control room unfiltered air inleakage rates that are commonly assumed in licensee's analyses of the control room habitability. The staff recently issued proposed generic communication (letter) on control room envelope habitability in Federal Register (May 9, 2002) for public comment (ADAMS Accession No. ML021090031). The staff also recently issued two draft regulatory guides: DG-1114, "Control Room Habitability at Light-Water Nuclear Pwer Reactors (ADAMS Accession No. ML020790125) and DG-1115, "Demonstrating Control Room Envelope Integrity at Light-Water Nuclear Power Reactors (ADAMS Accession No. ML020790191), for public comment.

Summarize the performance test results obtained in 1989 and state in detail how you estimated unfiltered air inleakage (200 cfm) using the system performance test results. Provide any additional substantiated bases subsequent to the test performed in 1989 that support 200 cfm unfiltered air inleakage rate you assumed. You should include, as appropriate, results from any subsequent inleakage and/or system flow tests performed, maintenance performed on the system to minimize the inleakage, and any modification/upgrade done to the system to improve the system integrity.

NMC Response

On February 28, 1989, Wisconsin Public Service Corp (WPSC) submitted an updated control room habitability-evaluation report to address NRC concerns over control room ventilation (CR Steinhardt (WPSC) to Document Control Desk (NRC). In this letter, WPSC concluded the following actions were appropriate:

1. Performance Characteristics needed to identify potential system improvements and/or updating the control room analysis will be identified.
2. These performance characteristics will be quantified through measurements.
3. The results will be reconciled with the analysis and procedures.
4. Finally, a review of the USAR and Technical Specifications will be made and changes or license amendments will be implemented accordingly.

The attachments to the February 28, 1989 submittal document the WPSC actions and response to the above actions. The response for action #1 indicated that maintenance and testing practices would be evaluated to periodically inspect that the boundary seals of the control room ventilation system are not degraded. Revisions to plant preventative maintenance procedures (PMP-25-1) were made to address this concern.

Action item 2 was completed by performing airflow tests. Fluor Daniel, and NUCON International performed an evaluation of the control room air-conditioning system performance. The test method was similar to that used in August 1986 by Mr. J. Hayes, NRC, and consultants from Argonne National Laboratory (ANL) while conducting a survey of the Kewaunee Nuclear Plant control room ventilation systems. The results of the survey were presented in a letter from Mr. M.B. Fairtile, NRC, to MR. D.C. Hintz, WPSC, dated January 29, 1987. The results were also incorporated in NUREG/CR 4960.

The NRC/ANL survey measured flow rates using a hot wire anemometer. Flow balances were performed to determine whether the habitability systems were performing within their design parameters. System airflow measurements were made to determine the unfiltered inleakage into the control room envelope and system performance. The results from the NRC/ANL survey were part of the basis of the NRC internal memo to the Regional Administrators from Mr. James Lieberman, Office of Enforcement, dated May, 28, 1987 regarding enforcement for control room habitability issues.

The testing performed by Fluor Daniel was the same as that done by NRC/ANL except that a Pitot tube and electronic micro-manometer were used instead of a hot wire anemometer. The Pitot tube was used to reduce the susceptibility of the reading being affected by turbulence. The Pitot tube measurement is not as susceptible to turbulence as is the hot wire anemometer. There was the potential that the hot wire anemometer would detect any air leaking into the duct through the test location. That air could be misinterpreted as leakage past the damper.

Unfiltered inleakage was determining by using the measured leakage through the closed dampers (48 CFM), allowance for leakage through building elements (80 CFM), and adding an assumed air exchange based on door opening and closing (10 CFM)(48 + 80 + 10 = 138 cfm). This leakage was adjusted for the worse case unfiltered inleakage resulting from one of the redundant dampers failing to close. This resulted in leakage through closed dampers (110 CFM), allowance for leakage through building elements (80 CFM), and adding an assumed air exchange based on door opening and closing (10 CFM). This resulted in a total unfiltered inleakage of 200 CFM (110 + 80 + 10 = 200CFM). For the dose to the operator evaluation, only one train of post-accident cleanup filtration was assumed to operate in-spite of the assumption of a failed damper.

Action item 3 was addressed in the February 28, 1989 submittal. Procedures (Operating and Maintenance) have been updated several times since the February 28, 1989 submittal. Design Change Request (DCR) 2373 added a redundant start signal to the post-accident ventilation system.

Action item 4 was addressed when a review of the USAR, Technical Specifications, and procedures was performed. Revisions to the USAR were made under the USAR Assessment Project. Technical Specifications were revised in accordance with the requirements of Generic Letter 99-02.

Maintenance performed on the system to help minimize inleakage has included replacing the caulking around the shield blocks on the north wall and installing new flashing. The door to the equipment room has also been replaced. In addition, routine preventative maintenance inspections include door inspections, barrier inspections, and ventilation boundary inspections.

NRC Question #3

Provide a figure showing the reactor containment vessel, shield building, auxiliary building, control room, control room normal and emergency air intakes, refueling water storage tank and all source term release points.

NMC Response

KNPP's "Updated Control Room Habitability Evaluation Report", dated February 1989, Figure 3 is a general layout of the intake locations. This figure contains general elevations of the Control Room, Auxiliary Building, and Containment. Also shown on this figure is the location of the intake air for the Control Room. See attachment 2, updated Figure 3.

KNPP's Shield Building contains the Reactor Building within the enclosure. The Refueling Water Storage Tank is located within a sub-compartment of the Auxiliary Building called Auxiliary Building Zone Special Ventilation.

The following summarizes the release points for each accident.

LOCA: Consideration is given for releases from general containment to the shield building, auxiliary building, and directly to the environment. Releases to the shield building are heldup, filtered, and released over time to the environment through the reactor and shield building exhaust stack. Releases to the auxiliary building may come from general containment leakage, leakage from systems containing ECCS recirculation water, and leakage to the RWST from the ECCS systems. All of these release points are in the auxiliary building zone special ventilation. This system maintains a vacuum and filtered discharges are made through the auxiliary building exhaust stack.

Steam Generator Tube Rupture: Consideration is given for primary coolant leakage to both the ruptured steam generator and the intact steam generator. To maximize release to the environment it is assumed the release is to the atmosphere via steam relief system (i.e., steam dump, power operated relief valves or safety valve). The steam exits through the relief system valves, which are located in the same general area on the east and west side of containment.

Locked Rotor: Consideration is given for primary coolant leakage into both steam generators. The radioactivity is conservatively assumed released to the atmosphere through the steam relief system (i.e., steam dump, power operated relief valves or safety valves).

Rod Ejection: Consideration is given for releases from general containment to the shield building, auxiliary building and directly to the environment. Releases to the shield building are heldup and released over time to the environment through the reactor and shield building exhaust stack. Releases to the auxiliary building may come from general containment leakage, leakage from systems containing ECCS recirculation water, and leakage to the RWST from the ECCS systems. All of these release points are in the auxiliary building zone special ventilation. This system maintains a vacuum and discharges are made via the Auxiliary Building exhaust stack.

In addition to the containment leakage, consideration is given for primary coolant leakage into both steam generators. The radioactivity is conservatively assumed released to the atmosphere through the steam relief system (i.e., steam dump, power operated relief valves or safety valves).

Fuel Handling Accident: Two accidents are taken into consideration. The first accident assumes a fuel assembly has been damaged in the containment building. The activity is released to the general containment area and discharged via the containment purge system to the atmosphere. The containment purge system is connected to the reactor and shield building exhaust stack.

The second accident assumes a fuel assembly has been damaged in the spent fuel pool handling area. Consistent with KNPP's Technical Specification 3.8, the Spent Fuel Pool Sweep System is in operation. The system will collect any radiation released and discharge is directed to the Auxiliary Building Exhaust Duct.

Steam Line Break: To maximize release to the environment the ruptured steam line is assumed to be outside containment. It is assumed that the release is to the atmosphere. To conservatively bound the analysis, it is assumed that the discharge is in the same location as the power operated relief valves and main steam safety valves. In addition to the faulted steam generator, the analysis also considers primary coolant leakage to the intact steam generator. The radioactivity is conservatively assumed released to the atmosphere via the steam relief system (i.e., steam dump, power operated relief valves or safety valves).

Gas Decay Tank Rupture and Volume Control Tank Rupture: These tanks are located in the auxiliary building zone special ventilation area. To maximize radiation release to the environment it is assumed that the radiation is discharged from these areas via the Auxiliary Building Ventilation Discharge Exhaust Duct.

NRC Question #4

In Section 2.2.2, "Containment Modeling" of Attachment 2 to your submittal (Attachment 2), you assumed that:

- during the first 10 minutes of the accident, 90 percent of activity leaking from the containment is discharged directly to the environment and 10 percent enters the auxiliary building,
- after 10 minutes, only 1 percent of the activity leaking from the containment is discharged directly to the environment, 10 percent continues to go to the auxiliary building, and the remaining 89 percent will go to the shield building, and once the shield building is brought to subatmospheric pressure at 30 minutes into the event, the iodine is subject to removal by recirculation through filters. In addition, you assumed various shield building air flow rates in Table 12 of Attachment 2.

Provide substantiated technical bases for these timing, release fractions, and air flow rates assumed stating why some of these parameters are different from those listed as the design bases for the radiological analyses in Section 14.3.5 of the Kewaunee USAR and in Attachment 3, "Updated Control Room Habitability Evaluation Report," to your letter dated February 28, 1989 (UCRHER).

NMC Response

The containment modeling reflects the licensing basis of KNPP. Leakage from the primary containment is assumed to be 0.5%/day for the first 24 hours. For the remainder of the 30 day period the leakage is assumed to be 0.25%/day. KNPP Technical Specification 6.20, Containment Leakage Rate Testing Program, states "The maximum allowable leakage rate (L_a) is 0.5 weight percent of the contained air per 24 hours at the peak test pressure (P_a) of 46 psig." This provides the basis for assuming 0.5%/day. In accordance with Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluation Design Basis Accidents At Nuclear Power Reactors" the leakage is decreased by 50% following the first 24 hour period. This provides the basis for assuming 0.25%/day after the first 24 hours.

The total allowable leakage is split into three different areas (i.e., auxiliary building special ventilation zone, shield building and environment). TS 6.20 states "For penetrations which extend into the Auxiliary Building Special Ventilation Zone, the combined leak rate from these penetrations shall not exceed $0.10L_a$ This provides the basis for the assumption that 10% of the total allowable leakage, L_a, enters the auxiliary building special ventilation zone. T.S. 6.20 also states that for penetrations which are exterior to both the shield building and the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.01L_a$."

This provides the basis for the assumption that 1.0% of the total allowable leakage, L_a, leaks directly to the environment. Based on the previous values (i.e., 10% and 1.0%) the assumed amount to the Shield Building is 89%. Based on this Technical Specification it is assumed that 10% of the leakage goes to the Auxiliary Building Special Ventilation Zone, 89% of the leakage goes to the Shield Building, and 1% of the leakage goes to the environment.

The radiological analyses that assume releases via containment are dependent on the performance of the Shield Building Ventilation System. Technical Specification 5.2, Containment, describes the function of this system. In general terms, the system is designed to produce a vacuum throughout the annulus. Once a vacuum is achieved, the system will circulate the air discharging only enough air to account for inleakage into the Shield Building. During the initial vacuum establishment period no credit is assumed for the shield building. Therefore, during the first 10 minutes the analysis assumes 90% of the activity leaking from containment goes to the environment (i.e., 1% + 89%). Following achievement of a vacuum in the shield building credit is taken for this compartment in the radiological analysis. For periods greater than 10 minutes containment leakage is divided as previously discussed (i.e., 1% to the environment, 10% to the auxiliary building special ventilation zone, and 89% to the shield building).

Three time periods are associated with the shield building ventilation (SBV). During the first period, 0 to 10 minutes, the SBV system starts and draws a vacuum in the shield building. During this period, no credit is taken for the shield building. The second period, 10 to 30 minutes, takes into consideration modulation of the SBV system dampers to maintain a vacuum during the initial events of the accident. During this period credit is taken for the filtration of the volume being discharged through the SBV system however, no credit is taken for recirculation. The final period (greater than 30 minutes) consists of stable system operation with a combination of recirculation and discharge to maintain the vacuum. These assumed time periods are different than previously submitted to the NRC (reference Updated Control Room Habitability Evaluation Report dated February 28, 1989). The intent of these changes is to allow for potential relaxation in the performance of the SBV system. This is made possible by the release timings associated with the alternative source term methodology.

NRC Question #5

Table 12 of Attachment 2 lists containment vessel volume as 1.32E6 ft^3. State the sprayed and unsprayed volumes of the containment vessel.

NMC Response

The containment vessel volume is stated as the net volume (i.e., total volume less structures and components) therefore 1.32E6 ft^3 is the sprayed volume. This is consistent with the licensing basis value used to determine Containment Spray capability in previous submittals to the NRC (reference Updated Control Room Habitability Evaluation Report dated February 28, 1989).

NRC Question #6

Table 14.3-8 of the USAR and the UCRHER list the fission product removal coefficients for elemental and particulate iodine for the containment vessel internal spray system as 10.0 and 0.45 per hour respectively. Contrary to these values, you proposed in this license amendment to use iodine removal coefficients of 20 and 5 per hour for iodine in elemental and particulate forms respectively. Explain the discrepancies in detail.

NMC Response

The iodine removal coefficients have been recalculated for this submittal. The values in KNPP's USAR Table 14.3-8 and KNPP's Updated Control Room Habitability Evaluation Report dated February 28, 1989 were conservatively calculated. There are no system or component changes associated with this change. The new analysis was performed by Westinghouse per NRC SRP 6.5.2.

An error in the Westinghouse report has recently been found. The report lists the assumed spray fall height as 65 feet (on pages 10 and 43). This is incorrect. A spray fall height of 150 feet was assumed in the calculation of the spray removal coefficients, consistent with the value assumed in the KNPP's Updated Control Room Habitability Evaluation Report.

NRC Question #7

State the basis for 0.91 hour switch-over time to recirculation spray from the start of the accident.

NMC Response

The shortest time to drain the RWST from the level corresponding to the Technical Specification minimum water volume to the 12% level setpoint for injection spray termination (10% setpoint plus 2% uncertainty) was calculated modeling one high, one low head (RHR) and one containment spray (CS) pump in-service; each delivering maximum calculated flow at zero reactor coolant system/containment pressure. The time was calculated to be 3298 seconds from the time the SI signal was generated. This was rounded down to 0.91 hours for use in the analysis.

At the 12% switchover level the operator stops the RHR and CS pump; therefore the above time represents the minimum time at which spray would be stopped. It is recognized that if more pumps were operating the draindown time would be shortened, but making that assumption is not consistent with either the assumption that the core becomes degraded or the assumption that spray removal of airborne activity is limited to that provided by one spray pump operating at its minimum flow rate. While this is not a limiting assumption for minimum spray time, it is an appropriate assumption for the accident analysis. The calculation is documented in Westinghouse calculation note CN-FSE-99-68, titled "Kewaunee Inputs to SGTR and LOCA Off-Site Dose Analysis". In order to support the timely review of the Kewaunee LAR, Westinghouse has transmitted this calculation to its Rockville office where it is available for viewing.

NRC Question #8

Tables 12 through 18 list the major parameters used in the radiological consequence analyses for the design-basis accidents. List the reactor power level and the duration of accident assumed for each design-basis accident. You stated in Section 1.2 of Attachment 2 that the fission product activities in Table 5 are increased by an additional 10 percent to cover future power uprate.

NMC Response

The core source term presented in Table 5 of the LAR was calculated using a core power level of 1683 MWt (1650 + 2%). The calculation was performed with ORIGEN2.1 (Reference 8-1) by modeling a 3-region core with an 18,700 MWD/MTU cycle burnup corresponding to a 550-day cycle. Representative power sharings were used in the ORIGEN2.1 for each region. The inventory of the 3 regions was summed to provide the core inventory.

The reactor coolant noble gas and alkali metal activities presented in Table 6 of the LAR and the gas decay tank and volume control tank activities presented in Table 19 of the LAR are also based on operation with a core power level of 1683 MWt, assuming one percent fuel defects. Reactor coolant concentrations are calculated assuming a one percent fuel defect level in the core and modeling the 550-day cycle with reactor coolant concentrations reduced by boron dilution and demineralizers. An ORIGEN2.1 core inventory is an input to the calculation. It is assumed that there is no purging of the volume control tank during the cycle. The maximum concentration of each nuclide is selected for reporting and providing input to additional calculations. However, for accident dose analysis, the reactor coolant iodine concentrations are based on operating at a limit of 1.0 μCi/gm Dose Equivalent I-131. Volume control tank inventory was taken from the calculation of reactor coolant activities. Gas decay tank activities were calculated by assuming the reactor coolant is degassed at the maximum letdown rate at reactor shutdown.

In performing the accident dose analyses, the core source term was increased by 10% to provide margin in the analyses to facilitate the evaluation of a future power uprate. (It is expected that the doses thus calculated will bound those that would be determined for the future uprate). The same 10% increase was used for the gas decay tank and volume control tank inventories and for the reactor coolant concentrations for noble gases and alkali metals. The 10% increase was not applied to the iodine concentrations since the reactor coolant iodine activity is based on the defined operating limit.

The secondary steam release data provided in Tables 13, 15,16 and 18 of the LAR show the values modeled in the analyses. These values were calculated assuming the nominal core power of 1650 MWt, increased by factors of 2% to 4.5% to account for uncertainties and to provide margin. In performing the accident dose analyses, the calculated steam releases were increased to provide margin in the analyses to facilitate the evaluation of a future power uprate.

The analyses presented in the LAR are not intended to support a power uprate, just to include margin to demonstrate the feasibility of an uprate from a dose standpoint. Final analyses in support of an uprate would determine whether the reported doses are bounding.

The power level modeled and the duration of activity releases for each of the design basis accidents are listed in Table 8-1.

Table 8-1

Event	Core Power Level For Source Term	Core Power Level for Secondary Steam Release	Duration of Releases
Large Break Loss Of Coolant Accident	1650 MWt + 2%	Not Applicable	30 days
Steam Generator Tube Rupture	1650 MWt + 2%	1650 MWt + 4.5%	8 hours
Locked Rotor	1650 MWt + 2%	1650 MWt + 4.5%	8 hours
Rod Ejection	1650 MWt + 2%	1650 MWt + 2%	30 days
Fuel Handling Accident	1650 MWt + 2%	Not Applicable	2 hours
Steam Line Break	1650 MWt + 2%	1650 MWt + 4.5%	72 hours
Gas Decay Tank Rupture	1650 MWt + 2%	Not Applicable	5 minutes
Volume Control Tank Rupture	1650 MWt + 2%	Not Applicable	5 minutes

Reference 8-1: RSIC Computer Code Collection CCC-371, "ORIGEN2.1: Isotope Generation and Depletion Code —Matrix Exponential Method", 2/96.

NRC Question #9

In Section 1.2 of Attachment 2, you stated that control room operator doses were determined for duration of the event. The staff request you recalculate control room operator doses for 30 days for all design basis accident as illustrated in your Reference No.11 of Attachment 2 independent of the fission product release duration. The airborne fission products intruded into the control room atmosphere may remain well after the fission product releases are terminated.

NMC Response

For control room dose calculation purposes, the duration of the event was assumed to continue beyond the time of release termination, until 30 days, to account for the continued exposure of the operators to activity in the control room.

NRC Question #10

In Section 2.2.4 of Attachment 2, you assumed that the emergency core cooling system (ECCS) leakage to the auxiliary building and the residual heat removal (RHR) back-leakage to the refueling water storage tank (RWST) are 6 gph and 3 gpm, respectively. State substantiated bases for these assumptions and where these limits are specified in the design-basis documents or in the plant operating procedures. State how you modeled the fission product transport and release through the RWST to the environment.

NMC Response

The emergency core cooling system (ECCS) leakage to the auxiliary building and the residual heat removal (RHR) back-leakage to the RWST are 6 gph and 3 gpm respectively. These values are consistent with values stated in "KNPP's System Integrity Plan." (Revision A dated April 13, 2000) This plan is concerned with leakage from systems outside containment that could contain highly radioactive fluids post-accident. Currently, performance of KNPP surveillance procedures ensures these leakage values are not violated through a combination of visual inspections and hydrostatic tests.

NRC Question #11

In Section 2.2.4 of Attachment 2, you also assumed that the iodine partition factor is reduced to 1 percent once the auxiliary building sump water temperature is below 212 F. Provide the technical bases to justify the lower iodine partition factor assumed.

NMC Response

KNPP's USAR Section 6.2.5, Effects of Leakage From Residual Heat Removal System, identifies that the temperature of the containment sump recirculation water is below 212°F when ECCS recirculation begins. Based on this, the analysis assumed conservatively that 1% of the iodine is released. This item is taken from the original Final Safety Analysis Report (FSAR) and is considered Licensing basis. The same justification can be made in the new analysis. In addition to being below 212°F, the ECCS back-leakage to the RWST is also being injected into a tank of water that will be significantly below 212°F. The leakage path back to the RWST is via the suction lines of the Safety Injection, Internal Containment Spray and Residual Heat Removal Pumps. These lines will be filled with water. Additionally, the RWST will have a height of water that will act as a cooling mechanism as the liquid enters the tank. Based on these items a decrease in the iodine partition factor was assumed.

NRC Question #12

In Section 2.2.4 of Attachment 2, you also assumed that half of the iodine activity that becomes airborne from two leak sources in the auxiliary building is removed by plateout on surfaces. Justify your assumption.

NMC Response

KNPP's USAR Section 6.2.5, Effects of Leakage From Residual Heat Removal System, identifies that the iodine released from spilled coolant would largely be plated out with approximately 50% within structures before release through the ventilation system. The area of the release is the same and therefore the plate out assumption was carried forward.

NRC Question #13

Table 1 of Attachment 2 shows the radiological consequences of the postulated design-basis accidents. List dose contributions from each fission product release pathway (containment leak, ECCS leak, and RWST back-leakage release) to the LOCA doses (EAB, LPZ and control room).

NMC Response

The answer to this question is still under development. NMC's response will be submitted at a later date.

NRC Question #14

List the control room atmospheric relative concentrations (χ/Q values) used in your control room operator dose calculations for each fission product release point. The χ/Q values shown in Table 4 bound all release points?

NMC Response

The calculation supporting the original Updated Control Room Habitability Evaluation Report dated February 28, 1989 used a bounding control room dispersion factor. The value reported in Table 4, of that report, bounds all release points.

NRC Question #15

In determining the radiological consequences resulting from the design basis steam generator tube rupture (SGTR) accident, provide the following information:

- letdown flow rate
- primary coolant mass
- iodine appearance rates for each iodine nuclide
- average iodine concentrations over 0 to 2 hours and 0 to 4 hours
- amounts of iodine and noble gas released over 0 to 2 hours and 0 to 4 hours

NMC Response

Letdown flow rate = 88 gpm (80 gpm +10% uncertainty) modeled with perfect cleanup. The calculation of the iodine appearance rates also considered 12-gpm leakage from the primary system.

Primary coolant mass = 268,474 lbm

Iodine appearance rates, including spike factor of 500:

I-131 = 148.6 Ci/min
I-132 = 354.0 Ci/min
I-133 = 253.9 Ci/min
I-134 = 159.7 Ci/min
I-135 = 187.6 Ci/min

In the analyses performed for Kewaunee, the coolant activity is calculated on a continuous basis, and average activities are not available. The average activity can be calculated as a simple average of the activity at the start and end of the interval. Table 15-1 provides the iodine concentrations in the RCS resulting from the accident-initiated iodine spike at the time periods requested. The analysis models the activity initially in the reactor coolant system separately from the activity that enters the reactor coolant system from the fuel due to the accident-initiated iodine spike. The total iodine concentration is not available for the time intervals requested. The data in Table 15-1 represents the activity in the RCS resulting from the spike.

Table 15-1

	Accident-Initiated Iodine Spike Activity in Reactor Coolant System at End of Time Period (Ci)		
	0 hours	2 hours	4 hours
I-131	0	1.671E+04	3.432E+04
I-132	0	3.039E+04	4.840E+04
I-133	0	2.776E+04	5.539E+04
I-134	0	9.350E+03	1.154E+04
I-135	0	1.919E+04	3.584E+04

The amounts of iodine and noble gas released to the atmosphere are provided in Table 15-2 for the pre-accident and accident-initiated iodine spike cases. The calculations did not include a separate determination of the total release at 4 hours, so the total activity released until 2 hours and until 8 hours is provided.

Table 15-2

	Pre-accident Iodine Spike Activity Released (Ci)		Accident-Initiated Iodine Spike Activity Released (Ci)	
	0 – 2 hours	0 – 8 hours	0 – 2 hours	0 – 8 hours
I-131	3.352E+02	3.361E+02	1.636E+02	1.677E+02
I-132	2.848E+02	2.850E+02	3.642E+02	3.670E+02
I-133	4.907E+02	4.917E+02	2.770E+02	2.830E+02
I-134	5.683E+01	5.684E+01	1.518E+02	1.521E+02
I-135	2.675E+02	2.679E+02	2.009E+02	2.041E+02
Kr-85m	9.292E+01	9.315E+01	9.292E+01	9.315E+01
Kr-85	4.414E+02	4.436E+02	4.414E+02	4.436E+02
Kr-87	5.596E+01	5.599E+01	5.596E+01	5.599E+01
Kr-88	1.720E+02	1.723E+02	1.720E+02	1.723E+02
Xe-131m	1.480E+02	1.487E+02	1.480E+02	1.487E+02
Xe-133m	2.323E+02	2.334E+02	2.323E+02	2.334E+02
Xe-133	1.262E+04	1.268E+04	1.262E+04	1.268E+04
Xe-135m	1.616E+01	1.616E+01	1.616E+01	1.616E+01
Xe-135	4.519E+02	4.535E+02	4.519E+02	4.535E+02
Xe-138	2.022E+01	2.022E+01	2.022E+01	2.022E+01

NRC Question #16

Show that the iodine gap activity would be depleted within 4.0 hours terminating iodine spike at that time for the postulated SGTR accident.

NMC Response

In order to maintain the reactor coolant system below the technical specification limits only a fraction of the fuel rods in the core can have defects that allow activity to leak from the gap. The fraction of fuel with leaking defects that corresponds to the analysis assumption for I-131 activity is 0.762% (i.e., cladding defects are present in fuel rods producing 0.762% of the core power). I-131 is used for this calculation since it is the most significant contributor to the dose due to its long half-life and high dose conversion factor. The total I-131 activity in the gap of these rods is determined using the core activity from Table 5 of the LAR (4.48E7 Ci) and the gap fraction from Table 3 of RG 1.183 (8%). The activity is increased by 10% to provide margin to allow for a future uprate. With these inputs the total activity to be released is 3.00E4 Ci. The spike appearance rate for I-131 is 148.6 Ci/min (provided in the response to Question 15). At this rate the gap activity of 3.00E4 Ci is released in 202 minutes. This is less than 3.4 hours. This was conservatively increased to 4.0 hours for use in the dose analysis.

NRC Question #17

Table 13 of Attachment 2 lists the bounding SGTR thermal hydraulic parameters for the radiological consequence analysis. Provide the references for the following parameters used:

- tube rupture break flow
- tube rupture break flow flashing fractions
- amounts of steam released to the environment from the ruptured and intact SGs
- termination of steam release from the intact SGs at 8 hours after the accident
- termination of steam release at 30 minutes from the faulted SG
- termination of steam release at 72 hours from the faulted SG after the main steam line break accident

NMC Response

The steam generator tube rupture thermal hydraulic parameters (e.g., break flow, flashing fractions, and steam releases) were calculated for the replacement steam generator program, and are presented in Table 6.2-2 of Reference 17-1.

Termination of releases from the ruptured steam generator at 30 minutes is consistent with the plant licensing basis and the USAR. Releases from the ruptured steam generator are stopped when the operators have isolated the ruptured steam generator and initiated the plant cooldown with the intact steam generator. This would occur before 30 minutes.

In the steam line break dose analysis, releases from the faulted steam generator are terminated at 72 hours. This time was chosen to conservatively bound the time required to bring the reactor coolant system temperature below 212°F. This is accomplished by first dumping steam from the intact steam generator until the RHR system can be brought into service, and then by the RHR system.

Reference 17-1: Westinghouse letter KEW-LIC-00-096, LTR-ESI-00-302, "Kewaunee RSG – Final Licensing Report Submittal", dated November 11, 2000, which transmitted Westinghouse report, "Kewaunee Nuclear Power Plant Steam Generator Replacement and T_{avg} Operating Window Program", dated November 2000.

NRC Question #18

List the control room isolation times for each design basis accident in a separate table with its bases and its initiating signals. State if you included the switchover time to the ventilation system after safety initiation signal.

NMC Response

The control room HVAC is switched from normal operation mode to emergency mode either on a safety injection (SI) signal or on an air duct high radiation signal. The signal to switch the control room HVAC mode is generated within 63 seconds of a SI signal and immediately by the high radiation signal. Once the signal is generated, the switchover is completed within 10 seconds.

The Large Break LOCA and Steamline Break accidents result in a safety Injection signal being generated almost immediately following the start of the event. The Rod Ejection and Steam Generator Tube Rupture events take some time to depressurize the reactor coolant system to the low pressurizer pressure SI setpoint at which time the SI signal is generated. The Locked Rotor, Fuel Handling Accident, Gas Decay Tank Rupture and Volume Control Tank Rupture events rely on the air supply duct radiation monitor signal for control room isolation. The high release rates and conservative transport assumptions associated with these events, as modeled in the analysis, result in the high radiation monitor action setpoint being reached almost instantaneously following the start of the release.

In all of the analyses the time assumed to initiate emergency mode HVAC is longer than the time calculated based on signal generation and appropriate delays.

Table 18-1 summarizes the information for the various accidents.

Table 18-1

Event	Initiating Signal For Control Room Emergency Mode HVAC Actuation	Time of Initiating Signal For Control Room Emergency Mode HVAC Actuation (sec)	Delay For Control Room Emergency Mode HVAC Actuation (sec)	Time Control Room Emergency Mode HVAC Is Credited (sec)
Large Break Loss Of Coolant Accident	Low Pressurizer Pressure SI	Immediate	< 73	120
Steam Generator Tube Rupture	Low Pressurizer Pressure SI	174	< 73	300
Locked Rotor	High Activity in Air Supply Duct	Immediate	< 10	60
Rod Ejection	Low Pressurizer Pressure SI	40	< 73	120
Fuel Handling Accident	High Activity in Air Supply Duct	Immediate	< 10	60
Steam Line Break	Low Steamline Pressure SI	Immediate	< 73	300
Gas Decay Tank Rupture	High Activity in Air Supply Duct	Immediate	< 10	30
Volume Control Tank Rupture	High Activity in Air Supply Duct	Immediate	< 10	30

NRC Question #19

Table 16 of Attachment 2 lists assumptions used for rod ejection dose analysis. Provide technical bases and/or references for the following parameters:

- fraction of fuel melting
- steam releases to the environment

NMC Response

The potential for fuel melting exists only if a fuel rod enters DNB. The analysis uses the conservative assumption that 15% of the fuel rods enter into DNB. Due to the highly-peaked radial and axial power distribution, less than 50% of the fuel rods in DNB are assumed to experience melting and the melting would extend over less than 50% of their active length. Analyses have been performed which demonstrate that at the worst (highest power) location in the core, the amount of melting is limited to less than the innermost 10% of the limiting fuel pellet. Using these inputs, the total fraction of fuel melting is given by 0.15 x 0.1 x 0.5 x 0.5 = 0.00375.

As described in the USAR, the design basis rod ejection transient results from a failure of a control rod mechanism pressure housing which results in a loss of coolant accident with a possible reactivity insertion event. The steam generator steam releases to the environment were chosen to bound those calculated for the small break loss of coolant accident (SBLOCA). A 2 in^2 break was used. This is smaller than the flow area that results from a control rod mechanism pressure housing failure. The smaller break conservatively extends the steam releases and has a slower primary depressurization, thus maximizing the time until SI actuation (on low pressurizer pressure). It also delays the time when the primary pressure drops below the secondary pressure and extends the time when the steam generators are steaming to remove decay heat.

The SBLOCA analysis shows the primary pressure dropping below the secondary pressure at around 800 seconds. To conservatively bound this the primary to secondary leakage and atmospheric steam releases were continued in the analysis until 1800 seconds (0.5 hour). A plot of primary and secondary pressure from the SBLOCA analysis is provided in Figure 19-1. A very conservative step function is used to bound the SBLOCA steam releases. An 800-lbm/sec flow is assumed from the start until 200 seconds and a flow of 100 lbm/sec is assumed from 200 seconds until 1800 seconds. Figure 19-2 presents the step function superimposed on the SBLOCA analysis calculated flow rates. [Note that the steam releases before reactor trip are not included since the flow would be passed through the condenser with a partition coefficient of 100 prior to trip and the assumed loss of offsite power. The conservative assumption of loss of offsite power at the start of the event and immediate initiation of steam releases at a high rate assure that the analysis is bounding.]

Figure 19-1

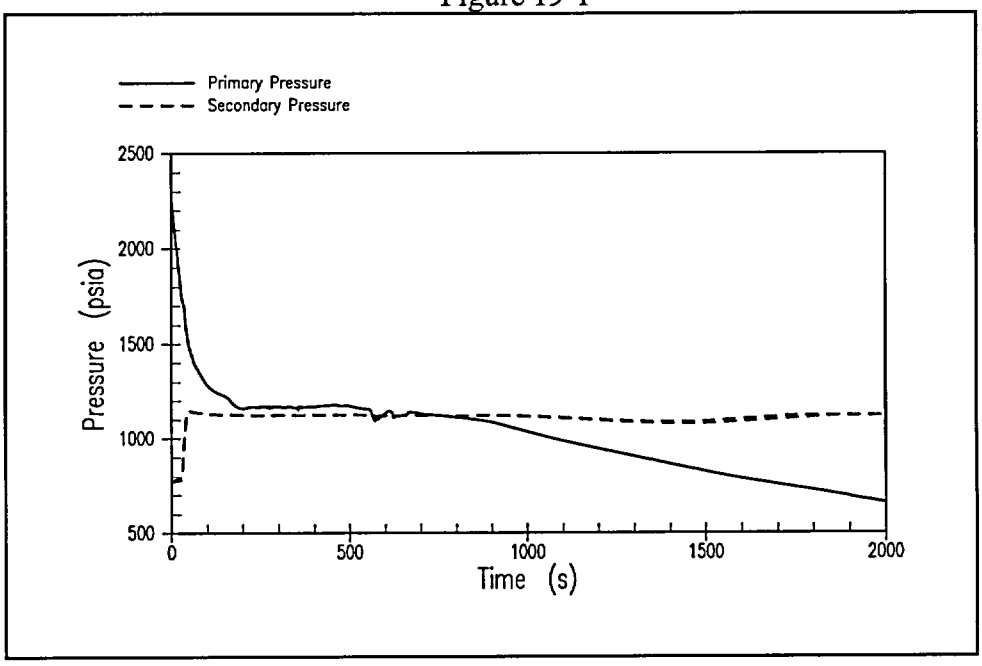
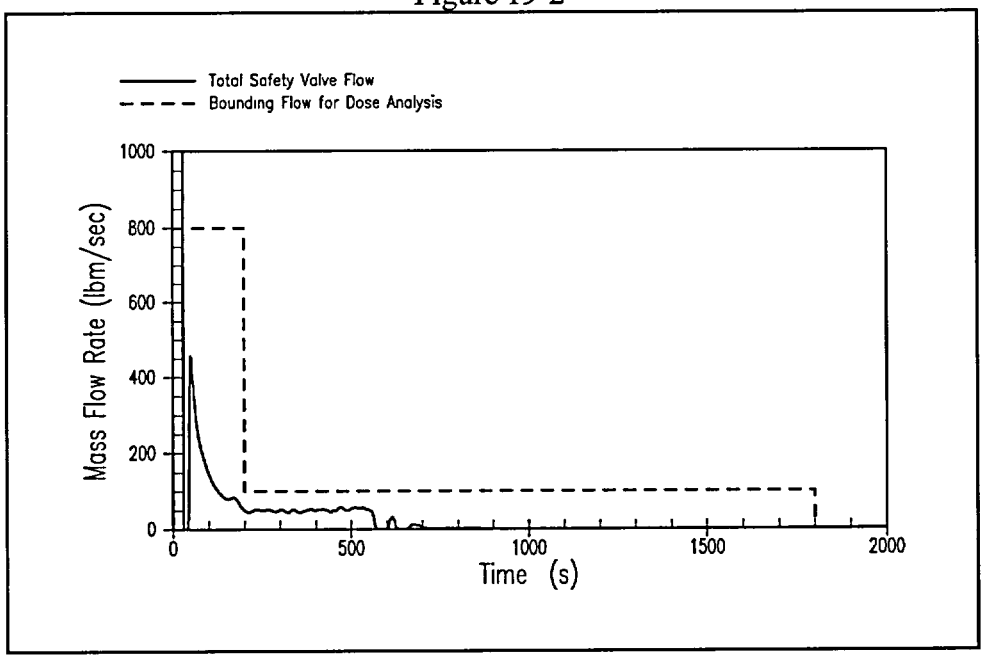


Figure 19-2



NRC Question #20

Table 17 of Attachment 2 lists assumptions used for fuel handling accident dose analysis. Provide technical bases for chemical forms assumed in release to the environment as70 percent in elemental form and 30 percent in organic form.

NMC Response

RG 1.183 specifies that the iodine leaving the pool is 57% elemental and 43% organic. This is based on a DF of 500 for elemental iodine and 1 for organic iodine. (i.e., total iodine leaving the pool = 99.85%/500 + 0.15%/1 = 0.2% + 0.15% = 0.35%. Of this 0.2/0.35 = 57% is elemental and 0.15/0.35 = 43% is organic.) However, the analysis was performed using the overall pool DF of 200 that is specified in RG 1.183. Based on an initial iodine species split of 99.85% elemental iodine and 0.15% organic iodine, the elemental DF that would result in an overall DF of 200 is approximately 286. With the overall DF of 200, 0.5% of the activity would leave the pool. (i.e., total iodine leaving the pool = 99.85%/286 + 0.15%/1 = 0.35% + 0.15% = 0.5%.) Of this, 0.35/0.5 = 70% is elemental and 0.15/0.5 = 30% is organic.

The split between elemental and organic iodine leaving the pool has no impact on the analysis since the control room filter efficiencies for the two iodine forms are the same, and no other filtration or removal processes are credited.

NRC Question #21

Table 19 of Attachment 2 lists the fission product source terms used for the gas decay tank and the volume control tank rupture dose analyses. Explain in detail why these source terms are differ from those listed in Tables D.6-1 and D.7-1 in Appendix D to Chapter 14 of the Kewaunee USAR (See pages 14.2-8 through 14.2-10). What are the bases for 5 minute release time while the Kewaunee USAR correctly assumed it to be an instantaneous release (see Section 14.2.3 of the Kewaunee USAR).

NMC Response

The first difference in values in Tables D.6-1 for the volume control tank and D.7-1 for the gas decay tank is due to the reactor coolant sources used as input for calculation of the sources. The USAR reactor coolant sources were generated for annual cycles whereas the Table 19 sources were generated for an 18-month cycle. In addition, the Table 19 sources were generated with ORIGEN2 which will result in some differences due to nuclide data compared to the USAR tables which were developed in the early 1970's. The basic method for calculating the reactor coolant sources in the USAR and in the LAR are similar - a calculation of activities based on 1% fuel defects in the RCS with coolant cleanup by demineralizers and deboration.

The table of specific activities for the volume control tank are normally part of the calculation of the reactor coolant sources. Differences between Table D.6-1 and the LAR Table 19 can result from the 18-month cycle and ORIGEN2 code used for the LAR Table 19. However, there appear to be other differences in the table which cannot be identified at this time.

The USAR D.7-1 table for the gas decay tank sources is generated by using Table D.4-1 of the USAR multiplying by reactor coolant volume of 6100 ft^3 with conversion of units. Hence, time elapsed for purging the RCS and the attendant nuclide decay are not included. For Table 19, the inventory of the gas decay tank following shutdown at end of cycle is calculated with degassing of the reactor coolant with letdown at the maximum rate. No purging of the VCT is assumed during the cycle resulting in an inventory of noble gases in the VCT vapor at end of cycle. For the iodines, the RCS concentration is assumed to be reduced by the resins and the remaining concentration in the VCT vapor is determined by a partition factor. For both the noble gases and iodines a calculation is done to simulate the purging of the VCT to the gas decay tank at shutdown, degassing of the RCS with the maximum letdown rate for 3 hours followed by another purge to the gas decay tank. The cycle of degassing for 3 hours and purging to the gas decay tank is continued for a total of 10 purges. Maximum concentrations over the degassing period are selected to make up the gas decay tank inventory.

The release timing has no impact on the calculated offsite doses (provided the release is completed within 2 hours) since all activity is released and no decay is modeled. The release timing impacts the control room doses, which are not currently included in the USAR. An instantaneous release would be inappropriately conservative for the control room dose calculations since it would allow all the activity to be transferred into the control room and then effectively lock it inside when the control room is isolated. For the fuel handling accident RG 1.183 specifies that the radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.

The tank ruptures are similar to the fuel handling accident in that there is a release into the auxiliary building and then a gradual transfer from the auxiliary building to the atmosphere. Based on this similarity to the fuel handling accident a 2-hour release period is appropriate, but 5 minutes was chosen since it comes closer to the typical assumption of an instantaneous release.

In addition, the gas decay tanks are located within the auxiliary building zone special ventilation area. The ventilation in the auxiliary building zone special ventilation area is drawn to and discharged out the auxiliary building discharge stack which is monitored via radiation monitors R-13 and R-14. High radiation due to a gas decay tank release would result in the auxiliary building zone special ventilation being actuated. This actuation causes a vacuum to be drawn in the Gas Decay Tank area and results in the ventilation discharge being directed through HEPA and Charcoal Filters prior to being released to the atmosphere. This would limit the amount of release occurring from this area. The gas decay tanks are also physically located in the basement of the Auxiliary building. A release would require the gas to be transported through the auxiliary building out the radiation -monitored ventilation ductwork and back into the control room.

Actuation of the auxiliary building zone special ventilation system combined with the physical location of the gas decay tanks in the auxiliary building basement provide further justification for a 5 minute release duration assumption for the gas decay tank rupture accident.

ATTACHMENT 2

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

September 13, 2002

Figure 3 Update

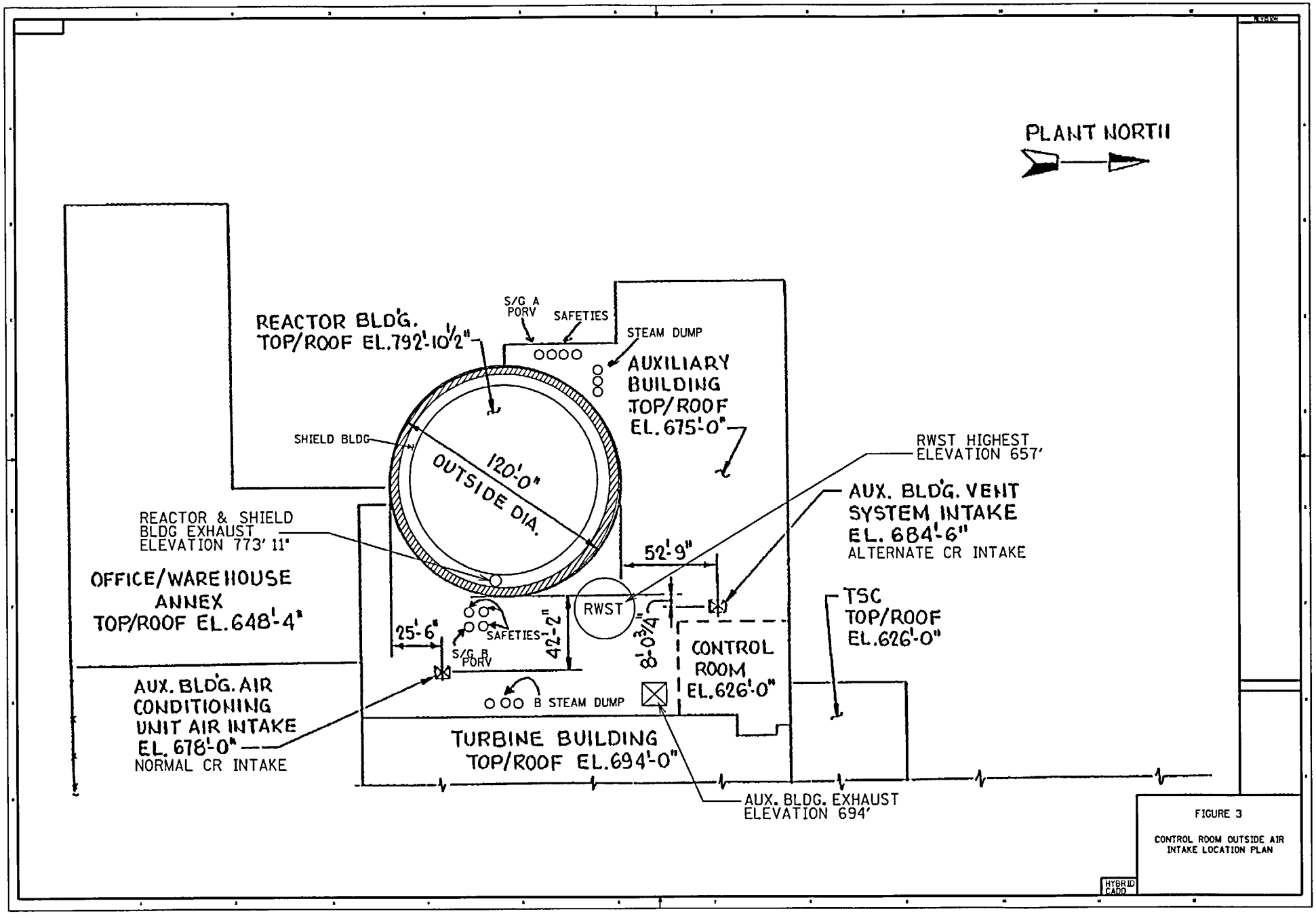


FIGURE 3

CONTROL ROOM OUTSIDE AIR INTAKE LOCATION PLAN