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SUBJECT: Forwards request for addl info for amend to secondary containment & standby gas treatment sys Tech Specs.

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December 4, 1997
GO2-97-218

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT TO SECONDARY CONTAINMENT AND STANDBY GAS TREATMENT SYSTEM TECHNICAL SPECIFICATIONS (ADDITIONAL INFORMATION)**

References:

1) Letter GO2-96-199, dated October 15, 1996, PR Bemis (SS) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

2) Letter, dated October 15, 1997, TG Colburn (NRC) to JV Parrish (SS), "Request for Additional Information for the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2) (TAC NO. M96928)"

3) Letter GO2-97-202, dated November 7, 1997, PR Bemis (SS) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment Technical Specifications - Schedule for Submittal of Additional Information"

Reference 1 included a request for amendment to the WNP-2 Technical Specifications for Secondary Containment and the Standby Gas Treatment (SGT) System. In Reference 2, additional information was requested on this subject. In accordance with the schedule provided in Reference 3, the requested additional information is attached.

A001

Should you have any questions or desire additional information pertaining to this letter, please call me or Mr. P.J. Inserra at (509) 377-4147.

Respectfully,

Paul Inserra
FOR
D.W. Coleman
Acting Manager, Regulatory Affairs
Mail Drop PE20

Attachment

cc: EW Merschoff - NRC RIV
KE Perkins, Jr. - NRC RIV, Walnut Creek Field Office
TG Colburn - NRR
C Poslusny, Jr. NRR
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DL Williams - BPA/399
PD Robinson - Winston & Strawn

Question 1: With the exceptions of secondary containment draw-down time, bypass leakage, mixing volumes, and SGT system flow rate addressed in the October 15, 1996 submittal, were assumptions other than those listed in Table 15.6-9 of the updated FSAR used in the evaluation of this request on the design basis LOCA analysis? If so, clearly state these assumptions and provide a technical justification for the change.

Response: Other major assumptions and data associated with the information in Table 15.6-9 are listed as follows:

- Primary containment is assumed to remain constant at the calculated peak internal pressure, P_a (38.0 psig) for the 30-day duration of the postulated LOCA, which keeps all release rates conservatively constant.

- No halogen plateout is assumed to occur in the secondary containment (Reactor Building) in accordance with Regulatory Guide 1.3, Revision 2.

- Control Room Emergency Filter Unit efficiency is unchanged from previous analysis at 95 percent for elemental halogens, 99 percent for particulate halogens and 95 percent for organic halogens; resulting in an overall Control Room Emergency Filter Unit efficiency of 95.2 percent for the Regulatory Guide 1.3, Revision 2, specified mix of halogen species (91% elemental x 95% + 4% organic x 95% + 5% particulate x 99% = 95.2%).

- No credit is taken for control room double doors (10 cfm infiltration leakage added) in accordance with SRP 6.4.

- The primary containment inerted free air volume is 3.447E+05 cubic feet in accordance with FSAR Table 3.2-7 (Plant Design Assessment Report for SRV and LOCA Loads).

- The secondary containment free air volume is 3.50E+06 cubic feet in accordance with FSAR Table 6.2-12 (Chapter 6.0). However, the model conservatively does not include the air space above the refueling floor. The resulting secondary containment volume involved in mixing is taken to be 2.90E+06 cubic feet. The mixing volume is 40 percent of this reduced value.

- Standby Gas Treatment filtration efficiency is reduced to 98.7 percent to account for the 14 scfm vortex bypass for the fans.

- Thyroid inhalation dose conversion factors were updated to the latest International Commission on Radiological Protection (ICRP) 30 values (See response to Question 2).

- Specific assumptions on Chi/Q and time dependence are provided in the responses to other questions in this response.

- No credit was taken for fission product removal in the suppression pool as described in NUREG-0800, Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System."

- Initial Primary Containment activity is listed in Table 1 based on ORIGEN-II calculations consistent with Regulatory Guide 1.3, Revision 2.

- Data on isotopic activity concentrations in containment and releases to the environment associated with Table 15.6-9 have also been updated.

Table 1
Initial Primary Containment Activity

Isotope	Curies
I-131	2.54E+07
I-132	3.63E+07
I-133	4.97E+07
I-134	5.43E+07
I-135	4.68E+07
Kr-83m	1.03E+07
Kr-85m	2.03E+07
Kr-85	9.26E+05
Kr-87	3.77E+07
Kr-88	5.28E+07
Kr-89	6.24E+07
Xe-131m	1.02E+06
Xe-133m	6.33E+06
Xe-133	1.99E+08
Xe-135m	4.10E+07
Xe-135	4.74E+07
Xe-137	1.73E+08
Xe-138	1.57E+08

Question 2: Provide a description of the PADD computer code used to calculate the LOCA consequences. Include the basic dose equations solved, the analytical method employed, and the dose conversion factors used.

Response: The Post-Accident Design Dose (PADD) computer code is a FORTRAN program that evaluates offsite and control room doses as a function of time given the Regulatory Guide 1.3, Revision 2, source term. It was written to automate the spreadsheet analysis formerly used to perform the dose calculations described in our Justification for Continued Operation (JCO) and previous meetings with NRC-NRR on this subject. Dose calculation equations and conversion factors from Regulatory Guide 1.109, Revision 1, are used to calculate dose consequences for the exposure pathways evaluated; updated by the more recent iodine inhalation thyroid dose conversion factors given in ICRP 30.

The PADD output includes the releases of radioactivity from the plant during postulated accidents. Using the calculated release rates, PADD determines the radioactivity entering the remote air intakes for the Control Room. The method of analysis for determining downwind concentrations used in the computer calculation is based on the atmospheric diffusion methodology presented in NUREG/CR-5055, "Atmospheric Diffusion for Control Room Habitability Assessments." The PADD code also provides a solution of the radiation transport, decay, and accumulation equations applicable to the volumes of interest at WNP-2. The program allows case studies for determination of variable sensitivity and maximum allowable values within the Control Room dose limits.

The PADD code analytically models the distribution of the significant isotopes in the Regulatory Guide 1.3 source term, which was determined using ORIGEN-II at power uprate conditions as currently described in the FSAR. One hundred percent of the core noble gases and 25 percent of the core inventory of halogens are divided by the equivalent primary containment air space volume to determine the concentration immediately available for transport from primary containment. Leakage from primary containment is modeled to occur at the proposed WNP-2 Technical Specification leakage limits. The source concentration in primary containment is tracked over the post-accident period for 30 days to account for in-growth, decay and depletion because of leakage from containment. Decay does not deplete the plume concentrations during transit from the Reactor Building to receptor locations. For the primary containment leakage into secondary containment, the source term that leaks is diluted by 40 percent of the specified secondary containment volume, as described in the answer to Question 3.

This airborne concentration is then filtered through SGTS and released out the Reactor Building Elevated Release Duct. No credit is taken for reducing concentrations by means of SGTS filtration until the Reactor Building internal pressure is drawn down to -0.25 inches water gauge (w.g.) to conservatively account for the possibility of additional building leakage as the building is drawn down. Based on the design basis draw-down calculations, the draw-down time is conservatively taken to be 20 minutes.

The responses to Questions 4 and 5 provide additional details on the atmospheric dispersion modeling implemented in the PADD code through use of applicable time-dependent Chi/Q values. Doses to Control Room operators are calculated using standard point-kernel shielding factors developed from the geometry at WNP-2. The doses are determined from inventories of each nuclide present for direct radiation shine from the Reactor Building and the Emergency Control Room Filter Units.

Doses are calculated using Regulatory Guide 1.109, Revision 1, dose conversion factors for immersion in the contaminated cloud that accumulates in the Control Room for both gamma and beta radiation, as well as inhalation of the contaminated air. Iodine inhalation dose conversion factors from ICRP 30 are used to calculate the inhalation doses as shown in Table 2. For the Exclusion Area Boundary and Low Population Zone, whole body doses from plume immersion and thyroid inhalation doses are also calculated using Regulatory Guide 1.109 parameters.

Table 2
ICRP 30 Iodine Dose Conversion Factors (DCF)

Isotope	DCF (rem/Curie inhaled)
I-131	1.08E+06
I-132	6.44E+03
I-133	1.80E+05
I-134	1.07E+03
I-135	3.13E+04

Question 3: What is the secondary containment forced-air recirculation rate used in determining the 40% mixing volume assumption?

Response: WNP-2 does have safety-related, forced-air recirculation room coolers in Reactor Building Emergency Core Cooling System (ECCS) pump and Motor Control Center (MCC) rooms. However, there are no safety-related general building recirculation fans in secondary containment. The 40 percent mixing volume assumption has been shown to be valid because of the passive design features at WNP-2. Mixing of the primary containment leakage in the secondary containment volume is accomplished through suction from the SGTS fans, natural convection, diffusion, and mixing in individual rooms surrounding primary containment. Flow paths through the building to the SGTS inlet include multiple doors, vents, and hatchways between floors.

The secondary containment volume involved in the mixing conservatively does not include the free air volume above the refueling floor on the 606 foot elevation. The 40 percent mixing assumption is substantiated by results of the secondary containment building modeling using the GOTHIC computer code. The use of GOTHIC for secondary containment analysis was benchmarked against the results of RELAP4 for various high energy line break analyses to provide assurance of its validity. For further detail pertaining to GOTHIC model comparison information refer to Attachment 5 of our submittal (Reference 1).

This model used twenty building nodes and established flow paths horizontally on each floor and vertically between floors in the building. It accounts for the factors that affect mixing including applicable heat loads and heat removal rates, fuel pool heatup and evaporation, emergency lighting loads, and physical air flow pathways. Case studies were conducted at three separate elevations and similar response characteristics were observed for the xenon gas concentrations at the SGTS.

The airborne concentrations of a tracer xenon gas at the intake to the SGTS was modeled as a function of time using the GOTHIC code. The slope of the concentration increase and the time of peak concentration from the GOTHIC results were comparable to the PADD calculation that assumed 40 percent mixing. This methodology was presented to the Staff in the pre-submittal meeting of February 6, 1995, as summarized in the NRC meeting notes, "Summary of Meeting on Post-Accident Containment Response," dated March 6, 1995. The methodology is also discussed in item (a) on page 3 of 7 of Attachment 2 of our submittal (Reference 1). The analysis substantiates that the 40 percent mixing assumption in the specified volume of secondary containment is valid for SGTS secondary containment building response to primary containment leakage.

Question 4: Provide a description of the fission product transport model used to calculate the offsite and control room LOCA doses. Give all volumes (nodes), transfer rates between nodes and removal rates (or DFs) between nodes. What DF was assumed for fission product removal in the suppression pool?

Response: The fission product transport model starts with definition of the post-LOCA source term. Using the guidance from Regulatory Guide 1.3, 100 percent of the core inventory is the basis for all releases. One hundred percent of the core noble gases and 25 percent of the core inventory of halogens are assumed to be released instantaneously to the primary containment air space. No credit for plateout or suppression pool scrubbing of the halogens is taken for this source term. This inventory of noble gases and halogens is divided by the equivalent containment inerted air space volume to determine the airborne concentration subject to leakage from primary containment. Two separate classes of leakage from primary containment were used: (a) leakage from containment penetrations into secondary containment (Reactor Building), and (b) bypass leakage directly out of the building. These two leakage sources are handled separately because they were modeled to leave the Reactor Building at different locations.

Leakage from primary to secondary containment is calculated to occur at the WNP-2 Technical Specification limit of 0.5 volume percent per day. This leakage is assumed to be distributed to secondary containment and will partially mix with Reactor Building air as it is drawn into the SGTS inlet on the 572 foot elevation. Forty percent by volume mixing is assumed for this leakage that is drawn into the SGTS. The contaminated air is released through the Reactor Building Elevated Release Duct after passing through the SGTS. Credit for removal of halogens by the SGTS filters is not taken until the building pressure is drawn down to -0.25 inches (w.g.). Vortex damper leakage around the SGTS filter units of 14 scfm is assumed, causing a reduction of the halogen filtration efficiency from 99 percent to 98.7 percent after the building pressure is drawn down to -0.25 inches (w.g.). In the conservative design basis case presented, it is assumed that it will take twenty minutes for the building to be drawn down to that negative pressure. The SGTS flow rate from secondary containment is modeled as 5000 cfm per train, which provides the basis for the minimum SGTS flow rate proposed in our submittal (Reference 1). For the design basis case, only one SGT train is assumed to be functioning.

Secondary containment bypass leakage is taken to be 18 scfh, vented directly to the outside. This value conservatively bounds the bypass leakage tests performed at WNP-2 to ensure penetration functionality. The leakage is conservatively assumed to occur from the Reactor Building wall closest to the Control Room, a distance of 39.8 meters from the local intake. For the remote intakes, the closest distance is 78.3 meters from the southeast corner of the Reactor Building.

The plumes of contaminated air are released to the outside atmosphere and are transported by means of atmospheric dispersion and diffusion in accordance with standard modeling described in more detail in response to Question 5. The dispersion of the concentrated plume is represented by the atmospheric dispersion factors, Chi/Q, that account for the effects of wind and dispersion based on site-specific meteorological parameters. The Chi/Q values used in this analysis are based on the WNP-2 physical configuration and 95 percent meteorological parameters as discussed in more detail in the response to Question 5. Chi/Q values are conservatively determined to envelop the actual conditions that could exist at the site at the time of an accident, and are specific to each of the two points of release and the air intake or receptor location being evaluated.

Contaminated air can be taken in by any of the three WNP-2 Control Room air intakes that may be exposed to the plume at the time of the accident. The local intake is located in the wall of the Radwaste Building above the Control Room, Remote Intake #1 is located 158 meters northwest and Remote Intake #2 is located 123 meters southeast of the Reactor Building Elevated Release Duct. In the LOCA event, the local intake isolates and the Control Room enters the pressurized, filtered mode of operation, maintaining a +1/8 inch w.g. pressure with respect to the atmosphere outside the Control Room. No credit can be taken for Control Room operator action to close one of the remote intake valves, if it is exposed to the plume, for two hours following the LOCA. It is not likely that both intakes would be exposed to significantly contaminated air at the same time, but during the course of the accident, the wind may shift and contaminate the other intake. Operator action to switch to an alternate remote intake will take about 30 minutes, and an additional 30 minutes is added for conservatism. The Control Room air intakes are all assumed to be contaminated for the first three hours following a LOCA, and for six hours each day after the first day to account for the plume wandering as the winds shift. It is assumed that the closed intake valves will leak and allow contaminated air into the intake.

The air taken into the Control Room intake is filtered through the Emergency Control Room Filter Units. The Technical Specification limit for bypass leakage around the filter unit is 0.55 cfm, and 10 cfm in-leakage is assumed to account for Control Room traffic through non-airlocked doors. No credit is taken for the Control Room double doors because the interlock feature is not used. The efficiency of the filter units is taken to be 95.2 percent overall for the mix of halogens specified. Doses to Control Room operators are calculated for the direct gamma contribution from build-up of contamination on the filters, direct gamma shine from the Reactor Building, gamma and beta doses from submersion in the cloud of contaminants that enter the Control Room, and inhalation of contaminated air that accumulates in the Control Room.

Table 3 summarizes the volumes (nodes) used in the model and Table 4 provides the transfer rates between nodes.

Table 3
List of Volumes

Containment Free Air Space	3.447E+05 ft^3
Secondary Containment Involved	2.905E+06 ft^3
Control Room	2.000E+05 ft^3

Table 4
Transfer Rates Between Nodes

Flow Path	Leakage Assumed	Filtration Effectiveness
Primary to Secondary Containment leakage	0.5 volume %/day	NA
Secondary Containment to atmosphere (filtration effective after 20 minute draw-down time)	5000 cfm (SGTS)	98.7%
Primary Containment direct to the environment (Secondary Containment bypass leakage)	18 scfh	0
Control Room filtered makeup air	1100 cfm	95.2%
Control Room unfiltered in-leakage	10.55 cfm	0

Doses to individuals who may be at the Exclusion Area Boundary or Low Population Zone are determined using the Chi/Q values calculated in accordance with Regulatory Guide 1.145, Revision 1, using the NRC PAVAN computer code, as described in WNP-2 FSAR Section 2.3.4, and Tables 2.3-34 and 2.3-36. The PAVAN code input consisted of meteorological data collected from 1984 to 1989 to develop the Chi/Q values. All releases are assumed to be ground level releases. The Chi/Q values used are based on Pasquill-Gifford with meander sigmas instead of the desert sigmas because they provide more conservative calculated consequences.

Question 5: What were the time-dependent Chi/Q values used for the control room air inlets? What Chi/Q values were used for control room in-leakage?

Response: Chi/Q calculations for determining conservative Control Room dose consequences are based on the model described in NUREG/CR-5055, *Atmospheric Diffusion for Control Room Habitability Assessments*, J.V. Ramsdell, Pacific Northwest Laboratory, May 1988. This NUREG provides justification for the new approach compared to the overly conservative Murphy/Campe Chi/Q calculation technique used in the past (13th AEC Air Cleaning Conference, *Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19*, K.G. Murphy and K.M. Campe). The atmospheric releases from WNP-2 are postulated to occur separately from: a) the SGTS effluent out of the Reactor Building Elevated Release Duct, which vents the secondary containment volume contaminated with primary containment leakage, and b) primary containment leakage that exits containment directly to the outside atmosphere (secondary containment bypass leakage). Based on the physical configuration of the Reactor Building Elevated Release Duct, the Reactor Building, the three Control Room outside air intakes, and parameters from the 95 percent meteorology at WNP-2, Chi/Q values are conservatively estimated for both contamination sources and each of the three Control Room intakes.

The NUREG/CR-5055 methodology allows taking credit for an elevated release for two-train SGTS operation discharging to the Reactor Building Elevated Release Duct. However, to conservatively envelop single train operation, the Composite Ground Release Model is used instead. This model gives a Chi/Q value of 9.65E–05 seconds/meter^3 for the distance (64.0 meters) from the Elevated Release Duct to the local Control Room intake. The closest remote intake (Remote Intake #2) is 123 meters from the Elevated Release Duct, giving a Chi/Q value of 4.40E–05 seconds/meter^3. This value is also conservatively applied to the furthest remote intake (Remote Intake #1), 158 meters from the Elevated Release Duct. This analysis used 95 percent meteorology, Stability Class "F", and a 0.75 meters/second wind speed for evaluating accident consequences. The value of k in the NUREG/CR-5055 model is taken to be 100 and the projected area of the Reactor Building, A, is 2883 meters^2.

Secondary Containment bypass leakage postulated to escape directly to the outside environment is modeled to occur from the Reactor Building wall closest to the Control Room local intake, a distance of 39.8 meters from the local intake, giving a Composite Ground Release Model Chi/Q value of 1.71E–04 seconds/meter^3. The distance to the closest remote intake (Remote Intake #2) is 78.3 meters, resulting in a Chi/Q value of 7.57E–05 seconds/meter^3, which is also conservatively applied in the calculation to the other remote intake.

To account for changes in wind speed and direction over time, after the initial three hours of the release, adjustment factors were applied to the Chi/Q values. These values are based on the wind direction correction factor in the Murphy/Campe paper of 0.75 + F/4 for eight to 24 hours after release initiation, and 0.50 + F/2 for one through four days after initiation, and F for four to 30 days post-LOCA, where F is the 95 percent meteorology wind direction frequency. For WNP-2, the 95 percent meteorology frequency for wind blowing toward the three sectors in the direction of Remote Intake #1 northwest of the Elevated Release Duct is 0.308. The frequency for wind blowing toward Remote Intake #2 (southeast of the Elevated Release Duct) is 0.0938. The larger of these values is used in the calculation for both remote intakes for conservatism.

In accordance with SRP 6.4, the Chi/Q values were reduced by a factor of four after three hours for control rooms with dual remote intakes with manual valve selection. In addition, the Murphy/Campe methodology allowed for adjustments for occupancy of 0.6 for days two through four, and 0.4 for days four through 30, in lieu of breathing rate adjustments, which are applied in the calculation to the Control Room Chi/Q values.

The Chi/Q values were applied to the three Control Room intakes in proportion to the flows into the Control Room from each intake. Both sources (elevated release duct and direct bypass leakage from primary containment to outside) are assumed to independently contribute to the Control Room intakes. That is, the consequences from each type of release are additive to determine total impacts on habitability. The Control Room in-leakage was assumed to occur from the local intake located above the Control Room at the Chi/Q values shown in Tables 5 and 6. The remote intake supplying the bulk of the air for the Control Room is assumed to be directly under the plume for the first three hours, and for six hours per day thereafter (the factor of four mentioned above from SRP 6.4). Control Room testing has shown that the operators can accomplish the valve realignment in less than 30 minutes when radiation monitors indicate that one of the intakes is drawing in contaminated air.

The initial three-hour delay in operator action is based on the mandated two-hour prohibition on Control Room operator action credit, the one-half hour to actuate the valves, plus an additional one-half hour for conservatism. The Control Room intake flow was calculated as 1000 cfm +10 percent, or 1100 cfm, with 300 cfm of the 1100 cfm input assumed to be from intake valve leakage from the intakes exposed to the plume at the time. The Control Room in-leakage was assumed to be 10.55 cfm, which includes the SRP 6.4 value of 10 cfm in-leakage for Control Rooms without airlock doors, and the 0.55 cfm Technical Specification limit for bypass flow around the Control Room Emergency Filtration Units.

The NUREG/CR-5055 methodology gives the results presented in Table 5 for the Control Room intake Chi/Q values. Table 6 summarizes the Chi/Q time dependent adjustment factor values used in this analysis. This was previously discussed in detail with the Staff, which resulted in general agreement with the approach described above [Record of Telecon between Messrs. Loren Sharp and Paul Macbeth of the Supply System and Mr. Jay Lee (NRC-NRR Shielding Analyst) on March 24, 1989].

Table 5
Control Room Intake Chi/Q Values

	Elevated Release Duct seconds/meter^3	Bypass Leakage from Reactor Building Wall seconds/meter^3
Local Intake	9.65E–05	1.71E–04
Remote Intake #2 (closest)	4.40E–05	7.57E–05
Remote Intake #1 (furthest)	4.40E–05	7.57E–05

Table 6
Chi/Q Time Dependent Adjustment Factor Values

	Time After Accident				
	0-3 hours	3-8 hours	8-24 hours	1-4 days	4-30 days
Occupancy	1.0	1.0	1.0	0.6	0.4
Wind Speed	1.0	1.0	1.0	1.0	1.0
Wind Direction	1.0	1.0	0.827	0.654	0.308
Dual Intakes with Manual Valves	1.0	0.25	0.25	0.25	0.25
Overall Reduction Factors	1.0	0.25	0.207	0.0981	0.0308