RELATED CORRESPONDENCE

6/26/81

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of ARIZONA PUBLIC SERVICE COMPANY, et al., (Palo Verde Nuclear Generating Station, Units 1, 2 and 3	DOCKET NOS. STN 50-528 50-529 50-530

6/26/81

INTERVENORS ANSWERS TO

APPLICANT'S FIRST SET OF INTERROGATORIES

GENERAL

Answer to Interrogatory No. 1.

The parties have agreed that the Intervenor may have an additional period of time within which to respond to the Request for Admissions. The date for said Response will be set by agreement of the parties.

Answer to Interrogatory No. 2.

See Answer to Interrogatory No. 1. above.

CONTENTION NO. 1.

Answer to Interrogatory No. 3.

The term "transfer factors" as used in Paragraph 1.a of the Explanation to Contention No. 1 means that fraction

DS03 S0/1

of a given quantity of radionuclide which remains in or on a certain environmental medium following dispersion by atmospheric processes. Transfer factors are used in computations of dose estimates to individuals near nuclear installations to predict the passage of individual nuclides from soil to plants. They are necessary to determine radiation exposure by ingestion of foodstuffs through various exposure pathways (such as the exhaust-air pathway for noble gasses and aerosols containing fission products).

Bruland, Franke, et al., used the following formula to describe the transfer of a radionuclide from the air at ground level to the edible part of a crop:

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$$f_i = V \cdot 86\,400 \cdot \left[\frac{r \cdot T_v \cdot F \cdot (1-e^{-\lambda_{Ei} \cdot t_e})}{\lambda_{Ei} \cdot Y_v} + \frac{B_{iv} \cdot (1-e^{-\lambda_i \cdot t_e})}{\lambda_i \cdot P} \right] \cdot e^{-\lambda_i \cdot t_h}$$

where f_i = transfer factor for nuclide i in pCi/kg fresh plant : pCi/m^3 air

V = total rate of deposition of the radionuclides by fallout and washout

r = retention factor, dimensionless

The retention factor indicates the fraction of the deposited radioactivity which is initially retained on the part of the vegetation that is above ground.

T_v = translocation factor, dimensionless, indicates the fraction of activity deposited on the leaves which enters the edible part of the plant in the course of time.

F = factor describing the loss of externally deposited activity during preparation of the plants in the kitchen, dimensionless.

λ_{Ei} = effective decay constant for isotope i in 1/d. λ_{Ei} is a measure of the rate at which superficially adsorbed radionuclides disappear from the surface of the plant by weathering processes and physical decay.

Y_v = vegetation density of the plants in kg fresh weight/m^2

t_e = exposure time in d
t_e is the time from sprouting of the plants until harvest.

B_{iv} = transport factor for a nuclide i in pCi/kg fresh plant : pCi/kg soil by root uptake from the soil.

P = 224 kg/m^2 mass of the earth in the plowed layer under one square meter of ground surface

λ_i = physical decay constant of a nuclide i in 1/d

$\lambda_i = \frac{\ln 2}{t_{1/2\,i}}$, where $t_{1/2\,i}$ is the half-life of the nuclide

t_B = reactor operating time

t_B is also the period of time in which radionuclides can accumulate in the soil

t_h = storage time of the products

f_A = see section 5.1.7.

Answer to Interrogatory No. 4.

Intervenor has not identified "each and every" transfer factor at this time. This Answer will be supplemented when all such transfer factors have been identified.

Answer to Interrogatory No. 5.

See Answer to Interrogatory No. 4 above.

Answer to Interrogatory No. 6.

The term "other data" was used generally and was not meant to connote a specific item of information or data in Regulatory Guide 1. 109. The Regulatory Guide contains a spectrum of information for predicting individual doses from various routine reactor emissions. Much of this information is based on educated calculations and speculative assumptions, given the difficulty of conducting actual experiments on reactor emissions. Also, since many of the parameters that determine dose from routine reactor emissions can only be estimated absent site-specific values, due to a lack of operational experience, the very nature of the analysis is best-guessing.

For example, the nuclide dispersion model or factor used to predict nuclide travel in air after release from a reactor stack is based largely on educated prediction of certain meteorological data, such as wind speed, wind direction, atmospheric moisture content, land topography, etc. These data can be measured over time for actual conditions during plant

operation, but there is no assurance that pre-operational meteorological conditions will mirror those during operation. This is to be expected and is a situation not new to scientific inquiry.

Answer to Interrogatory No. 7.

See Answer to Interrogatory No. 6 above.

Answer to Interrogatory No. 8.

Yes.

Answer to Interrogatory No. 9.

Intervenor has not completed a review of the data at this time; the Answer to this Interrogatory will be supplemented.

Answer to Interrogatory No. 10.

See Answer to Interrogatory No. 9 above.

Answer to Interrogatory No. 11.

Given the significance of transport factors in determining radiation dose to humans through the exhaust-air-plant pathway and plant-animal pathway as discussed in the Answer to Interrogatory No. 3 above, it stands to reason that if a given transport factor is low, the corresponding dose will be low. That is, if .5% of a given nuclide is said to be incorporated into a plant through its root system and into its leaves (which later are consumed directly by a human or by a human through cow's milk or meat) then the dose to that human

will be lower than it would have been if a full 1% of the given nuclide was said to have been incorporated into the plant. Accordingly, if several transport factors are under-estimated or low, the cumulative effect is to underestimate the dose accordingly. It also should be noted that Regulatory Guide 1.109 lists transport factors for these different path-ways, but only for stable elements. Given that both noble gases and fission products released in gaseous effluents are undergoing constant decay for varying lengths of time, and that each new nuclide, because of its chemical makeup, may transfer more or less readily to a plant, it becomes easier to see how the eventual dose estimates could be underestimated.

While the factors which may lead to an underestimation of dose are apparent, there remains substantial uncertainty as to the conditions under which the doses will be underestimated. As stated above in previous Answers, the Intervenor has re-quested certain information from the Applicant and until such time as that information is received, the Intervenor has no way of determining what those conditions are, when they may occur, and how.

Answer to Interrogatory No. 12.

See Answer to Interrogatory No. 11 above.

Answer to Interrogatory No. 13.

The Applicant's Environmental Report -- Operating License Stage, Chapter 5, Section 2 and Appendix 5B, assumes that

all gaseous effluent releases under normal operation will enter the atmosphere at ground level. However, in the real operation of PVNGS such releases will occur from stacks and exhaust vents at varying heights above the ground. Since wind patterns are known to be different at varying heights above the ground, exclusive use of the ground-level release model presents an unrealistic scenario of the points at which such effluents will be released. Consequently, the dispersion factor for such effluents will not reflect reality. Exclusive use of a ground-level release model will tend to concentrate gaseous effluent releases in a smaller area nearer the plant, when in reality, such releases from the elevated stacks will disperse over wider areas which may be farther from the plant. Accordingly, effluents scattered over wider areas will deposit in agricultural areas to the east and northeast creating the potential for uptake by plants in those areas. If such wider dispersion is not taken into account in the calculations, the resulting dose to humans from pathway exposures will be underestimated. Thus, the absence of analysis for elevated release points provides a basis for Contention No. 1.

Answer to Interrogatory No. 14.

Yes.

Answer to Interrogatory No. 15.

The Intervenor suspects that the absence of analysis for elevated releases will have an effect on the design objectives of 5 millirems annually to the whole body and 15

millirems annually to the skin and any organ. Which design objectives may be exceeded cannot be determined on the basis of the information presented in the ER-OL. It stands to reason that the most conservative analyses, those which attempt to meet the criteria of "as low as reasonably achieveable," would use models which most accurately reflect reality. Even Regulatory Guide 1.109 contains formulas for computation of "gamma dose from noble gas releases from free-standing stacks more than 80 meters high."

Answer to Interrogatory No. 16.

No. (At the time of the filing of this Contention.)

Answer to Interrogatory No. 17.

The projected annual release of noble gasses of 2,300 Curies was taken from Applicant's Final Safety Analysis Report, Vol. 9, Section II.3, Table 11.3-6.

Answer to Interrogatory Nos. 18 through 22.

The following is a list of Combustion Engineering reactors for which information is available and their noble gas releases (in Curies) for the year 1977:

Plant		Noble Gasses (in Curies)
Calvert Cliffs	1	23,000.0
Fort Calhoun		3,810.0
Maine Yankee		286.0
Millstone Point	2	2,280.0
Palisades		59.9
St. Lucie		25,400.0

(Taken from USNRC, Office of Management and Program Analysis, "Radioactive Materials Released from Nuclear Power Plants," Annual Report 1977, NUREG-0521, published January, 1979, Table 4.)

All reactors listed above are less than 850 MW and thus are not considered "large plants." CE has no prolonged experience in the U.S. with reactors larger than 1,000 MW and has never built 1270-MW reactors other than the PVNGS models. The very nature of their size will affect their operation. The 1270-MW units will have a larger core, greater water flow through it, larger steam generator piping and greater peak pressures-equipment and factors which will increase the chance for deviations from normal operating conditions. According to Komanoff, CE reactors in the first part of 1980 had the lowest capacity factor rating of the vendor's history. Komanoff attributed this decline in operating efficiency to such problems as steam generator tube denting and leakage.

The same factors which will make CE reactors operate less efficiently also produce the same result in all large reactors (those of 1,000 MW or larger). Komanoff has shown that the industrywide capacity factor for large reactors was about 51% in 1979. This figure has generally been considered to be representative of the operating efficiency of all large reactors and casts doubt on the Applicant's projected 75% capacity factor for PVNGS.

Of the 28 PWRs listed in the above-referenced NUREG-0521, 16 (in 1977) exceeded the 2,300 Curies of noble gases projected to be released from PVNGS. Those large reactors which exceeded that level and the number of Curies released are shown below:

Plant	Vendor/MW	Noble Gasses (in Curies)
Cook	Westinghouse/1,090	3,800.0
Trojan	" /1,130	3,070.0
Zion *	" /1,050	32,200.0

*The facility is listed in Table 4 of the above-referenced NUREG-0521 as "Zion 1 & 2,"--which implies that the total release of noble gasses is cumulative for two plants.

Answer to Interrogatory No. 23.

See Table 4, NUREG-0521. The Table lists noble gas releases (in Curies) for each plant for every year from 1970 through 1977, inclusive.

Answer to Interrogatory No. 24.

The 1977 data for Curie releases of noble gasses from operating CE reactors gives an average of 9,140 Curies per unit. Intervenor believes this figure to be a more realistic projection of releases from PVNGS than the Applicant's projected 2,300 Curies per unit. This Answer is not based on any other study, calculation or analysis.

Answer to Interrogatory No. 25.

Intervenor admits that no actinide or transuranic

isotopes are expected to be released from PVNGS during normal operations.

CONTENTION NO. 5

Answer to Interrogatory No. 26.

Subsequent to the filing of Contention No. 5, Intervenor has discovered new information regarding EPA's requirements for the 91st Avenue plant. Les Watson, Acting Assistant Superintendent for the 91st Avenue plant has told Intervenor that as of April, 1981, the EPA requirements have been met at the 91st Avenue plant. As a result of the discovery of this new information, the Intervenor would like to withdraw Paragraph 3 of Contention No. 5.

Answer to Interrogatory No. 27.

See Answer to Interrogatory No. 26 above.

Answer to Interrogatory No. 28.

Intervenor cannot answer this question until discovery is completed.

Answer to Interrogatory No. 29.

No. (At the time of the filing of the Contention). Additional studies, calculations or analyses, however, obtained through discovery may support the Contention.

CONTENTION NO. 6B

Answer to Interrogatory No. 30.

Yes.

Answer to Interrogatory No. 31.

The Nuclear Regulatory Commission (NRC) has proposed

measures to reduce the occurrences and mitigate the consequences of ATWS events in NUREG-0460, Vols. 1-4. In Vols. 3 and 4, NRC proposed four categories of alternatives for three classes of reactors. The preferred alternative in Vol. 3 was Alternative No. 3. Upon further investigation, the NRC determined that a greater degree of safety ("defense in depth") was needed to protect the public from the consequences of an ATWS event. As a result, the NRC in Vol. 4 chose as its preferred mitigation method a combination of Alternatives Nos. 3A and 4A (also referred to as Alternative No. 3½) to be implemented over different periods of time and by different administrative actions. The Staff proposed implementing the provisions of Alternative No. 3A by Order with some level of voluntary cooperation on the part of industry during the phase of equipment installation. Additionally, the NRC Staff submitted to the NRC for proposed rulemaking, implementation of Alternative No. 4A. The Advisory Committee on Reactor Safeguards (ACRS) also has provided the NRC with input. NRC action on the proposed rules for ATWS is expected shortly. (See NUREG-0606, Vol. 3, No. 1, Feb. 13, 1981, p. 20-21.) The Palo Verde Nuclear Generating Station (PVNGS) would fall into the category of "new operating plants and plants in the later stage of construction" (those with construction permits issued prior to Jan. 1, 1978). Should the NRC adopt rules and regulations requiring features and methods to prevent and mitigate the occurrence and consequences

of an ATWS event, and should the NRC Staff issue its proposed Order, PVNGS would be expected to comply.

Answer to Interrogatory No. 32.

Yes.

Answer to Interrogatory No. 33.

Resolution of ATWS as an Unresolved Safety Issue (USI) is of high priority in the best judgment of both the NRC and the industry. The Intervenor also believes resolution of ATWS is a necessity, considering the ease with which an ATWS event took place in June, 1980, at Browns Ferry Unit 3. The NRC Staff's position as stated in NUREG-0460, Vol. 4, is clear -- an acceptable level of safety for Combustion Engineering (CE) plants will not be reached until compliance with at least the provisions of Alternative No. 3A, and ultimately, pending NRC action and additional reveiw and study, with the provisions of Alternative No. 4A.

However, the safe operation of a nuclear power reactor, including PVNGS, is not determined simply by resolution of ATWS or any other USI. Pending a site-specific analysis of ATWS-mitigating measures for CE plants like PVNGS, as reccommended in NUREG-0460, Vol., 4 Appendix A, the Intervenor has no way of determining if PVNGS will be safer even with the incorporation of hardware and the performance of model analyses as outlined in Alternatives Nos. 3A and 4A. The safety of PVNGS is not solely dependent on a resolution of ATWS as an USI.

Answer to Interrogatory No. 34.

Yes.

Answer to Interrogatory No. 35.

Absent a site-specific analysis of ATWS-mitigating measures at PVNGS, the Intervenor has no way of determining if such measures will or will not bring the risk of an ATWS event at PVNGS to an acceptable level. The acceptable level cannot be determined by the Intervenor until (1) the above-mentioned site-specific analysis is performed, (2) mitigative and/or preventive measures (hardware and analyses) are detailed by the Joint Applicants, and (3) the Intervenor is made aware of what those measures may be.

Answer to Interrogatory No. 36.

Yes.

Answer to Interrogatory No. 37.

It is the Intervenor's position that specific features to mitigate the consequences of an ATWS event should be incorporated into the PVNGS design, but again, for the reasons stated above, the Intervenor has no way of determining what those specific features should be. The Intervenor assumes that those features potentially may include some or all of the hardware and analyses detailed in Alternative Nos. 3A and 4A for CE reactors, and may even include such measures that are not included in the two alternatives but may be necessary to mitigate the consequences of an ATWS event.

Answer to Interrogatory No. 38.

USNRC, Atomic Licensing Appeals Board, ALAB-444 (Nov. 23, 1977, In the Matter of the River Bend Nuclear Project.)

Answer to Interrogatory No. 39.

USNRC, NUREG-0460, Vol. 4, Appendix E, Tables E-3 and E-4; Hagen, Nuclear Safety, p. 199, (For frequencies greater than 10^{-5} reactor years.)

USNRC, SECY-80409, Sept. 4, 1980. (For frequencies between 10^{-5} and 10^{-4} reactor years.)

Answer to Interrogatory No. 40.

The lack of operating experience for large CE reactors leaves substantial uncertainty as to how the 1270-MW models at PVNGS will perform. Already there is some evidence of a degradation of CE capacity factors due to steam generator tube denting and leaks. Industrywide capacity factors for all large reactors have declined steadily over the past few years, averaging only about 51% in 1979.

For large reactors, a number of factors may contribute to a greater frequency of transients that require "scram" initiation or response. Among these are the larger size of the core, the greater water flow through it, and the larger number of control rods. The latter factor is most important considering the NRC Staff's view that CE's analysis of the effect of vessel-lifting on control rod deformation was incomplete and/or inadequate in several areas. It stands to reason that the larger

the number of control rods, the greater the potential for deformation under peak-ATWS pressures (3800 psi to 4100 psi in larger reactors) during vessel-head lifting.

Answer to Interrogatory No. 41.

No. See Answer to Interrogatory No. 39 above; plus ACRS Advice and Comments Report on ATWS proposals of NUREG-0460, Vol. 4, submitted to NRC, April 16, 1980.

CONTENTION NO. 7.

Answer to Interrogatory No. 42.

Contention No. 7 is based upon Joint Applicant's inadequate treatment of decommissioning costs in their ER-OL. Cost estimates provided in the ER-OL are inadequately calculated due to a lack of operational experience in decommissioning a plant of this size. In addition, the dismantling operations outlined in Section 5.8.1 of the ER-OL are overly simplified and lack specificity.

Answer to Interrogatory No. 43.

Intervenor maintains that the Joint Applicants have "inadequately figured decommissioning costs" based on their calculation of $57 million as the entire cost of decommissioning PVNGS. See Answer to Interrogatory No. 45.

Answer to Interrogatory No. 44.

Yes.

Answer to Interrogatory No. 45.

NRC Reg. Guide 1.86 describes four alternatives for retirement of nuclear reactor facilities which are considered acceptable by the NRC. The Joint Applicants in Section 5.8.1 of the ER-OL discuss dismantling as the method they will use to decommission the plant. The Joint Applicants then state an estimated cost of $57 million per reactor unit. A detailed engineering cost estimate for decommissioning a commercial power reactor was presented in 1975 testimony before the Pennsylvania Public Utility Commission concerning TMI Unit 1. At that hearing, a witness on behalf of General Public Utilities Services Corp., estimated the capital costs of dismantling to be on the order of $118 million.

A study prepared by Northeast Utilities on decommissioning costs for the Millstone 3 Nuclear Power Plant in Connecticut estimated a total cost of $264 million for decommissioning.

In addition to capital costs of decommissioning, a Pacific Gas and Electric study in 1976 estimated an additional cost of surveillance and light maintenance of between $60,000 and $330,000 annually.

Answer to Interrogatory No. 46.

Yes.

Answer to Interrogatory No. 47.

The methodology used by the Joint Applicants in Section 5.8.1 of the ER-01 is inadequate in its entirety. Numerous questions remain concerning the specifics of dismantling procedures.

Answer to Interrogatory No. 48.

See Answer to Interrogatory No. 45.

Answer to Interrogatory No. 49.

See Answer to Interrogatory No. 45.

Answer to Interrogatory No. 50.

Presumably the Joint Applicants will recover these costs through rates.

Answer to Interrogatory No. 51.

See Answer to Interrogatory No. 53.

Answer to Interrogatory No. 52.

See Answer to Interrogatory No. 53.

Answer to Interrogatory No. 53.

It is the Intervenor's position that the utility commissions of the various states involved in PVNGS could conclude that the decommissioning expenses were imprudently incurred and prohibit the Joint Applicants from recovering such expenses in rates.

Answer to Interrogatory No. 54.

See Answer to Interrogatory No. 53.

Answer to Interrogatory No. 55.

No.

Answer to Interrogatory No. 56.

Not applicable.

CONTENTION NO. 8.

Answer to Interrogatory No. 57.

A concrete slump test is an indicator of water/cement ratio, ambient air temperature, air content, cement temperature, and consistency of cement prior to pouring.

Answer to Interrogatory No. 58.

The information obtained from a concrete slump test includes the water/cement ratio, the ambient air temperature, the cement temperature and the consistency of the cement.

Answer to Interrogatory No. 59.

A concrete slump test measures the amount of water and air in the premixed cement.

Answer to Interrogatory No. 60.

When performing a concrete slump test, the premixed cement is poured into a 12" high by 6" wide metal cone or tube. The cone is then removed from the cement and the slump is measured.

Answer to Interrogatory No. 61.

If the slump is not of correct proportions, the cement will not meet its designed strength specifications.

Answer to Interrogatory No. 62.

The lab numbers, date of reports, placement numbers, ticket numbers, and any other means of identification will have to be supplied by Engineering Testing Laboratories, the Bechtel Corporation, or the Joint Applicants.

Answer to Interrogatory Nos. 63 through 65.

Object on the ground that the interrogatory calls for information which is irrelevant, immaterial, and not designed to lead to the discovery of admissible evidence.

Answer to Interrogatory No. 66.

The concrete slump test is an important indicator of the strength, integrity, and job specification proportions of the concrete which will support a system essential to reactor operation.

WITNESSES

Answer to Interrogatory No. 67.

Intervenor has not determined at this time who will be called as witnesses.

Answer to Interrogatory No. 68.

See Answer to Interrogatory No. 67 above.

Answer to Interrogatory No. 69.

See list of documents attached.

Answer to Interrogatory No. 70.

Intervenor has not determined at this time which exhibits will be used.

Answer to Interrogatory No. 71.

See Answer to Interrogatory No. 70 above.

DATED this 26th day of June, 1981.

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "Notice of Correction" have been served upon the following listed persons by deposit in the United States mail, properly addressed and with postage prepaid, this 26th day of Nov, 1981.

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