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Docket No. 50-255

August 26, 1969

J. McGough, DRL

Report to ACRS

CONSUMERS POWER COMPANY OF MICHIGAN

PALISADES PLANT

U.S. Atomic Energy Commission
Division of Reactor Licensing

Declassified per 8/8/84 memo fm J.L. Funches, NRC, to L.B. Scattolini, PDR. Copy attached.

ABSTRACT

The applicant has proposed to omit the reactor thermal shield from the Palisades facility. After reviewing all factors which could be influenced by omission of the thermal shield, we have concluded that we have no objection to the applicant's proposal.

The decision on whether post-accident iodine cleanup equipment is required in the Palisades Plant was deferred during the construction permit review to enable collection of onsite meteorology data. We and our consultants have reviewed the meteorology data obtained. Using the atmospheric diffusion factor which we think is justified by these data, we calculate offsite doses in excess of Part 100 guideline values and have concluded that some form of iodine cleanup equipment must be provided in the Palisades Plant.

INTRODUCTION

This report on the Consumers Power Company of Michigan, Palisades Plant presents to the Committee a discussion of two subjects to be resolved in connection with the Committee's operating license review of this facility. The first subject pertains to the applicant's proposed omission of the reactor thermal shield. An early indication of whether or not the thermal shield can be omitted is desired since the reactor internals are scheduled to be installed in the very near future. The second subject pertains to the need for a post-accident iodine cleanup system. A final decision on this matter is desired as soon as possible so that the applicant can make firm plans to order long lead-time equipment, should this be necessary.

1.0 PROPOSED OMISSION OF THE REACTOR THERMAL SHIELD

The applicant's proposal to omit the reactor thermal shield from the Palisades facility is set forth in Amendment No. 13 to the application, along with a discussion of each of the factors which could be influenced by omission of the thermal shield. Additional information on this subject was obtained at an ACRS Subcommittee meeting at the plant site on July 31, 1969, and at a meeting with the applicant on August 15, 1969. Certain critical information obtained at the latter meeting will be submitted in an amendment to the application prior to the September ACRS meeting.

The applicant's decision to omit the thermal shield from the Palisades design is an outgrowth of a Combustion Engineering (CE) study to determine what could be done if difficulties due to excessive vibration should be experienced in the Palisades reactor. Although difficulty was not expected, advance consideration was given because thermal shield vibration has been experienced in several operating PWRs. If vibration were experienced, the preferred solution identified by the applicant for the Palisades reactor, as was the case for two PWRs which actually experienced such problems* would be to remove the reactor thermal shield. This naturally raised the question of whether it is advisable to install the thermal shield in the first place. On the basis of this analysis, the applicant concluded that the thermal shield is not a necessary component in the Palisades reactor and that not installing it actually leads to a more satisfactory overall reactor design.

* AEC Safety Review (and Supplements I, II and III, thereto) of the Proposed Repair and Modification of the SENA and SELNI Reactor Internals, February 28, 1969.

The words "not installing" are used deliberately because the thermal shield for the Palisades reactor has been completely fabricated and is currently located on the reactor floor within the containment building, awaiting a final decision on whether it is to be installed or not. If not, it will have to be cut up in order to remove it from the containment building.

A. Reactor Vessel Irradiation Effects

The term "thermal shield" as used today is a misnomer. It was originally used because this shield was provided primarily for the purpose of reducing thermal stresses in the walls of the reactor vessel due to gamma heating. Thermal stresses due to gamma heating are relatively unimportant in current large water power reactors, and the principal function of the thermal shield is to minimize damage to the reactor vessel material due to fast neutron irradiation.

The reactor vendor for the Palisades Plant, Combustion Engineering (CE) has calculated the integrated fast neutron flux ($E > 1$ Mev) over the 40 year design life of the vessel <u>without</u> the thermal shield to be 3.64×10^{19} nvt. It is interesting to note that the comparable value calculated by Westinghouse for the recently approved Ginna plant <u>with</u> a thermal shield is 3.7×10^{19} nvt.

We have reviewed in considerable detail the calculational technique used by CE, which utilizes the P3MG1 code in cylindrical geometry, using 55 neutron energy groups, followed by a point-kernel integration to correct for power distribution and non-cylindrical geometric effects (e.g. corner elements). The calculational technique appears to us to be at least as good as that currently being used by the other major reactor vendors. The mesh spacing used for the point-kernel computation is quite fine ($\sim$ 1 cm) in the important regions near the core boundary. CE has checked the calculation technique against experimental fluence measurements in the Dresden, Shippingport and Yankee-Rowe reactors, (page II-I of Amendment 13). The calculations show good agreement with these experimental results and indications are that the calculational technique gives conservative results (higher than measured fluence) by from 4 to 10%. It is our present opinion that the CE calculational technique provides results within ± 30% of the true fluence.

To determine the increase in the nil ductility transition temperature (NDTT) associated with this fluence, CE uses a design curve which appears to be an envelope of "all available" data for ASTM type A302B, A302B Modified and A533B steels irradiated at 550°F. Most of the experimental data were obtained by NRL.

The Palisades vessel material is A302B - Modified, which is the terminology that was used for A533B material before the ASTM issued the A533B specification. Essentially the only difference between A302B and A302B - Modified or A533B material is the addition of 0.40 to 0.70% nickel to improve the impact properties of the material. There are, however, data that indicate that the impact properties of irradiated A302B Modified or A533B material can be either better or worse than A302B material, depending upon the amount of residual copper in the material. For the amount of copper in the Palisades vessel (0.25% max), these data indicate that the impact properties of the material will be close to, but still within the envelope of the CE design curve.

The unirradiated NDTT of the plate material used in the beltline region of the Palisades vessel was determined by drop weight tests to be a maximum of -30°F. The applicant maintains that this value for the plate material is also representative of weld metal and the heat-affected-zone (HAZ). This is based on comparision of Charpy impact test results at +10°F on plate, weld and HAZ materials, obtained when the welds used in the Palisades vessel were qualified. This comparision shows that the unirradiated impact properties of weld and HAZ metal are at least as good as those of plate material.

This deduction may not be valid in all cases. The potential variance should not, however be a significant consideration in this case because the Palisades materials surveillance program includes specimens of each of these materials, so that the item of real interest, the irradiated impact properties of each type, will be determined periodically. In addition, the applicant informed us that unirradiated Charpy specimens of each type of material (plate, weld and HAZ) are being stored and will be tested along with the first irradiated specimens.

The following table presents a summary of the initial NDTT, NDTT shift and NDTT at end-of-life (EOL) predicted by the applicant with and without the thermal shield in place. Note that because of the 70°F difference between the initial NDTT assumed in the PSAR and subsequently measured values on Palisades vessel material, the EOL NDTT with the thermal shield indicated in the PSAR is higher than the current value without the shield.

	With Thermal Shield (PSAR Values)	With Thermal Shield (current values)	Without Thermal Shield (current values)
Max Fluence	1.9×10^{-19}nvt	1.9×10^{-19}nvt	3.64×10^{-19}nvt
Initial NDTT	+40°F *	-30°F	-30°F
NDTT Shift	212°F **	220°F	262°F
NDTT (EOL)	252°F	190°F	232°F

* An assumed, conservative value for early design (PSAR) purposes.

** From an earlier design curve, The current curve is based on additional experimental data obtained since the PSAR.

The initial NDTT +60°F and the predicted NDTT shift curve discussed above are used to establish the initial pressure-temperature limits for primary coolant system operation. These limits will be revised, as necessary, when data from the materials surveillance program become available. It is these pressure-temperature operating limits and the minimum allowable impact properties to preclude failure of the vessel due to thermal shock that will determine if and when the vessel must be annealed. This consideration is discussed further below.

B. Surveillance Program

The materials surveillance program proposed for the Palisades Plant as described in the FSAR (pages 4-39), will not be changed whether the thermal shield is omitted or not. We have reviewed this program and found that it meets or exceeds all our current requirements. The program includes 6 vessel capsules (we currently require a minimum of 5 for this type of plant) plus 2 accelerated exposure and 2 thermal-exposure-only capsules (we have no requirements for such specimens). The entire program is in accordance with ASTM E-185-66, "Recommended Practice For Surveillance Tests on Structural Materials in Nuclear Reactors".

The exposure locations and number and type of specimens in each capsule are, in our opinion, satisfactory. The vessel capsules are retained within baskets which are welded directly to the vessel wall. We have determined that sufficient "archive" material (extra material from which the vessel was fabricated) is being retained to enable preparing sufficient specimens to load at least two additional capsules. The proposed specimen withdrawal schedule appears reasonable, and the six capsules provide sufficient specimens for at least one, and possibly two annealings.

C. Vessel Annealing

Based on the properties of the reactor vessel material (type 302-B Modified steel) and the calculated neutron fluence at the vessel wall, the applicant believes that the reactor vessel material will perform adequately for the life of the plant (40 years) with the thermal shield removed. However, if irradiation damage (embrittlement) should reach a level which would inhibit or prevent continued operation of the reactor, the applicant considers in-place annealing of the Palisades reactor vessel as a possible means of restoring ductility.

Although a step-by-step procedure has not been developed, the logistics and general mechanical problems related to in-place annealing have been considered. Based on these considerations, the results from the in-place anneal of the SM-1A vessel, and the results from experiments conducted by the U.S. Naval Research Laboratory (NRL-M-1753), the applicant has concluded that the Palisades reactor vessel could be annealed in place, using pump heat, at a temperature of about 650°F. Experimental data from tests using type A-302-B Modified steel, irradiated at 550°F to a fast fluence of 1 x 10^{19} and annealed at 640° to 650°F for 168 hours, show that a recovery of 30% to 50% of the original properties is possible. (NRL-M-1753)

The applicant would use the surveillance specimens to determine in advance when damage to the vessel material would reach a level requiring the vessel to be annealed. Surveillance specimens would also be used to establish the correct annealing time and temperature as well as the effectiveness of the actual anneal of the vessel.

We agree with the applicant that an in-place anneal of the reactor vessel could be performed if this should ever become necessary.

D. Effect on Reactor Vessel Stress and Primary Shield Heating

Radiation heating in the reactor vessel wall, with and without the thermal shield, has been reviewed. The applicant calculates a maximum temperature difference, radially thorugh the wall or axially along the wall in the core region, of less than 5°F with the thermal shield and less than 20°F without the thermal shield. The effect of this temperature gradient on reactor vessel stresses has been evaluated and the combined effects including cyclic stresses are well within stress intensity and fatigue limits with or without the thermal shield installed.

Removal of the thermal shield increases the heat deposited in the primary concrete shield walls surrounding the reactor vessel. The applicant calculates a heat load to the primary shield of 120,000 Btu/hr for normal plant operation with the thermal shield, and 126,000Btu/hr without the thermal shield. Ample capacity (180,000 Btu/hr) has been provided in the primary shield cooling system to accommodate the relatively small increase in heat load associated with omission of the thermal shield.

E. Hydraulic Effects and Vibration Monitoring

The removal of the thermal shield from the reactor vessel results in an increase in the free flow area in the annular downcomer formed by the vessel wall and the core support barrel. This increase in flow area (~44%) will result in a corresponding decrease in flow velocity.

Based on the results from a number of flow model test configurations, the applicant states that the flow distribution skirt previously added to the Palisades design tends to correct any flow maldistribution in the annular flow region. Tests are scheduled, using an air flow model of the Palisades design with the flow distribution skirt, to confirm that removing the thermal shield will not adversely affect core flow distribution.

The applicant's analysis shows that removal of the thermal shield has a negligible effect on the natural frequency of the core support barrel system, and consequently, the system with or without the thermal shield should have similar response to similar excitations. The removal of the thermal shield, however, should reduce the magnitude of the excitation by reducing the flow velocity and should eliminate the possibility of additional forces being transmitted to the core barrel by the thermal shield as the result of either forced or self-excited vibratory motions.

A preoperational test program is planned to detect possible vibration of the reactor vessel internals prior to reactor startup. The proposed vibration measurement program without the thermal shield installed is a modification of the applicant's original program and is not a result of removing the thermal shield. Eight uniaxial accelerometers will be located on the core support barrel (5 at the inlet nozzle elevation and 3 at the lower snubber elevation) to detect "ring" and cantilever motion of the barrel and to provide information on dominant excitation and natural frequencies of the core barrel structure. Test runs will be made with the plant under hot and cold conditions both before and after fuel loading. The data will be analyzed and a correlation between experimental and analytical results will be determined. The proposed instrumentation will not be available following the start of normal plant operation.

The proposed vibration monitoring program lacks one significant element — the ability to monitor for changes which might occur during the life of the plant. We have concluded that the vibration monitoring program is essentially independent of the present question on removing the thermal shield and need not be resolved now. However, we do plan to determine to what extent the applicant's program should be augmented to include provisions for monitoring the primary coolant system for changes in its characteristic vibrations during the life of the plant.

F. Effect on Potential Consequences of Accidents

In reviewing the possible effects that removing the thermal shield might have on the potential consequences of accidents, the loss-of-coolant accident was identified as the only one in which there might be significant effects.

As previously discussed, removing the thermal shield will increase the irradiation damage to the material of the reactor vessel wall. The effect of this damage as related to the operation of the emergency core cooling system must be considered. The applicant has reviewed this effect and based on the Combustion Engineering thermal transient--fracture mechanics analysis, concludes that for a fast fluence in the range of 3.5 to 4.0 x $10^{19} n/cm^2$, a crack depth of 30% to 40% of the wall thickness can be tolerated, concurrent with ECCS operation, without propagation of the crack through the vessel wall.

Our review of this thermal transient and its effect on the reactor vessel wall is still in progress. Developments in this area, as presented by all reactor manufacturers, are still being evaluated. We have determined, however, that since means have been provided for early detection of irradiation damage to the reactor vessel material, the effect of removing the thermal shield as related to the potential consequences of the LOCA is acceptable.

The applicant has also reviewed the maximum pressure differences and resulting stresses on critical reactor internals which are associated with the LOCA. Calculations show a slight increase in the pressure differentials (0 to $<$ 20%) as a result of removing the thermal shield. We have concluded that the slight increase in pressure differential calculated for the reactor vessel internals will not have a significant effect on the potential consequences associated with this accident.

G. Summary - Conclusion

Analytical predictions indicate that the Palisades vessel can operate for the full plant lifetime without the thermal shield in place and not experience irradiation effects that would be likely to jeopardize its safety. A satisfactory material surveillance program will provide experimental data on actual vessel material which will confirm or deny this prediction. In the event results from this surveillance program indicate irradiation effects are more severe than those predicted, the vessel can be annealed to correct materials properties sufficiently to enable continued safe operation. Omission of the thermal shield has the beneficial effect of reducing the possibility of excessive vibration of reactor vessel internals. On the basis of these considerations, we have concluded that we have no objection to the applicant's proposal to omit the reactor thermal shield from the Palisades Plant.

The applicant has indicated to us that he only wants "oral assurance" from the Committee that it has no objections to omission of the thermal shield.

2.0 NEED FOR POST ACCIDENT IODINE CLEANUP EQUIPMENT

The question of whether or not the Palisades Plant needs post-accident iodine cleanup equipment goes back to the construction permit review of this plant. At the construction permit review, the applicant proposed that meteorology data taken at its Big Rock Plant demonstrated that diffusion climatology on the eastern shore of Lake Michigan was so favorable that part 100 guidelines would be met in the event of an accident without iodine cleanup equipment. Neither we nor our meteorological consultants, ESSA, believed that the Big Rock data supported this proposition. At the time, we calculated potential two-hour doses at the exclusion area boundary due to a DBA of about 1050 rem to the thyroid, whereas the applicant calculated less than 300 rem.

The applicant agreed to make provisions for installation of iodine cleanup equipment late in construction and to obtain onsite meteorological data of sufficient extent to either prove or disprove their claims about lake shore diffusion climatology, and therefore the need for iodine cleanup equipment. On this basis, we and the ACRS agreed that the decision could be deferred until the operating license review. These agreements were made a matter of record in the ACRS letter, the staff Safety Evaluation and Consumers Summary of Application to the Hearing Board.

Amendment No. 9, consisting of Consumer's application for an operating license and the FSAR for the Palisades Plant, includes a very brief summary of the onsite meteorology data obtained and indicates that these data support Consumers' original claims about lake shore diffusion climatology. After review of the information contained in this amendment, we informed the applicant that the meteorology data presented were not sufficiently detailed nor adequately analyzed for us to determine whether we could agree with this proposition. The applicant therefore submitted additional detailed meteorology data in Amendment No. 12 to his application.

In Amendment 12 the applicant also reduced the containment design leak rate from 0.2%/day to 0.1%/day. As a result of this change the potential accident doses, as calculated by the applicant, were reduced to 122 rem (vs 285 rem in Amendment 9).

The meteorology data submitted in Amendment 12 include two months (September 1967 and February 1968) of joint frequency of occurence of wind direction and the standard deviation of wind direction (σ_θ). These data were collected at two locations onsite, one on the shoreline, the other inland behind the sand dunes. Neither we nor our consultant, ESSA (see Appendix A) consider the inland data station to be representative of inland diffusion or transport from the shoreline plant location because of the sheltered location of the data station.

The applicant calculated standard deviations of wind directions (σ_θ) from the wind-direction range data, using the relationship that the standard deviation is equal to the range divided by 4.3. He bases this on an interpretation of a statement in "Meteorology and Atomic Energy".(1) As acknowledged by this same reference, there are differing opinions as to what the correct value of this factor should be. Measured values for this factor range from about 4 to 9, and the 4.3 value used by the applicant is close to the lowest, least conservative extreme.

A specific study of the relationship between wind-direction range and σ_θ based upon measurements at a number of different sites using a variety of meteorological instruments was made by E. H. Markee [2]. We and our consultant, ESSA, (see Appendix B) have reviewed this document and a number of others which pertain to this subject [3, 4, 5 & 6] and have concluded that the ratio of 4.3 proposed by the applicant is not adequately conservative and that the more generally accepted value of 6.0 should be used.

The effect of using a higher value for this factor is that poorer conditions for atmospheric diffusion are predicted to occur at the site a larger fraction of the time. The significance of this is discussed further below.

Another problem with the applicant's analysis of his meteorology data has to do with his categorization of Pasquill stability conditions. On page D-10b of Amendment 12 the applicant indicates the generally accepted ranges of σ_θ values associated with each Pasquill stability category (Type). However, when he attempts to quantify the probability of each Pasquill stability category, wind speed and wind direction at the Palisades site, (page D-10c of Amendment 12), he changes the values of σ_θ associated with each Pasquill stability category. Pasquill Type F is generally represented by $\sigma_\theta < 3.7^\circ$, however, in his joint frequency summaries, (Tables 11 & 12 of Amendment 12) the applicant uses a value of $\sigma_\theta < 2.5^\circ$

to identify Pasquill Type F.

Based upon this latter categorization, and a factor of 4.3 relationship between wind-direction range and σ_θ , the applicant estimates that Type F stability conditions with 2 m/sec or less onshore winds will occur less than 3.2 percent of the time at the site and Type E stability conditions and 2 m/sec diffusion conditions or poorer, approximately 4.7 percent of the time. We generally take the position that the meteorological conditions used in calculating the two-hour exclusion-boundary doses should have a probability of occurring at the site no more than about 5 percent of the time, that is, better meteorological diffusion conditions than those used for the dose calculations should occur at least 95 percent of the time. The applicant, therefore, proposes that meteorogical conditions for Pasquill Type E and 2 m/sec wind speed are appropriate for the Palisades facility.

Using the generally accepted categorization for Type F conditions of $\sigma_\theta < 3.7^\circ$ and the 6.0 relationship between the wind-direction range and σ_θ , we and our consultants estimate that Type F and 2 m/sec or less wind speed conditions will occur at the Palisades site as much as 6.6 percent of the time. Even if we use the 4.3 relationship between wind-direction range and σ_θ proposed by the applicant but the generally accepted $\sigma_\theta <$ 3.7° categorization for Type F conditions, we estimate that Type F and 2 m/sec or less wind speed conditions still occur about 4.7 percent of the time.

It is significant to note that the meteorological parameters which are used on most power reactor facilities are those for Pasquill Type F conditions and 1 m/sec wind speed. Less conservative conditions than Type F and 1 m/sec conditions have been justified and approved in only a few cases (e.g., Turkey Point, Cook and San Onofre). For the D.C. Cook plant, in particular, which is located on Lake Michigan only a few miles from Palisades, Type F and 2 m/sec conditions were proposed and accepted.

In summary, we and ESSA believe that the applicant's onsite meteorology data justify the use of Type F meteorology with a wind speed of 2 m/sec rather than Type E and 2 m/sec as interpreted by the applicant. We have attempted to reconcile the difference in interpretation of the meteorology data by telephone conversations and meetings between our meteorology specialists and consultants and the applicant's but have not been successful. Using the diffusion factor associated with Type F meteorology and 2 m/sec ($X/Q = 2.6 \times 10^{-4}$), and our standard containment pressure decay model, we calculate a 2 hour exclusion boundary dose of 340 rem. The applicant calculates a corresponding dose of 122 rem, (Amendment 12, page 14.22-6).

We have reached the conclusion that some form of iodine cleanup equipment should be installed in the Palisades Plant in order to reduce the potential accident dose significantly below the 10 CFR 100 guideline value. Since the applicant has not been responsive to our informal suggestions on this, we plan to inform him of our conclusion in writing. Prior to doing so, however, we would like to consider any comments the Committee may have on this matter.

In connection with this problem, it is significant to note that all plants since Connecticut Yankee, with the exception of Palisades, have provisions for iodine cleanup of one form or another.

References

(1) Slade, David H. "Meteorology and Atomic Energy," USAEC of Tuh. Info, p. 275, July, 1968.

(2) Markee, E. H., Jr. "On the Relationships of Range to Standard Deviation of Wind Fluctuations," Monthly Weather Rev. 91(2) pp. 83-87, 1963

(3) (J. Z. Holland), U. S. Weather Bureau, A Meteorological Survey of the Oak Ridge Area, ORO-99, U. S. Atomic Energy Commission, Oak Ridge, Tenn., p. 584, November, 1953.

(4) Bowne, N. E., and R. R. Soller. "A Meteorological Survey of the CANEL Site at Middletown, Connecticut", U. S. Weather Bureau, Washington, D. C., p. 27, July, 1958.

(5) Pack, D. H., C. R. Hoseler, and T. B. Harris. "A Meteorological Survey of the PWR Site at Shippingport, Pennsylvania", U. S. Weather Bureau, Washington, D. C., p. 62, December, 1957.

(6) (G. A. McMarrais and N. F. Islitzer), U. S. Weather Bureau, Diffusion Climatology of the National Reactor Testing Station, IDO-12015, U. S. Atomic Energy Commission, Idaho Falls, Idaho, p. 149, April, 1960.

APPENDIX A

Comments on

Palisades Plant
Consumers Power Company
Final Safety Analysis Report
Volumes I, II, and III dated November 1, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 3, 1969

In addition to comments on the Final Safety Analysis Report, comments as a result of a visit to the site by ESSA personnel on January 30, 1969 accompanied by AEC personnel and representatives of the applicant are included. Of particular interest was the exposure of the shoreline and the inland wind measuring sites.

The shoreline exposure is on a telephone pole at a height of 200 feet above lake level about 700 feet from the shoreline and about 300 feet (2 containment building diameters) behind the reactor dome. Although this was a good exposure before the dome was constructed, it is now in the lee of the building for direct onshore winds. The inland exposure is, in fact, as stated in the report "at a sheltered location in the lee of a sand dune". Less than a few hundred feet away trees and vegetation tower over the 55-ft height of the anemometer. On the day of our visit, a pronounced onshore flow was felt and measured at the shoreline, but calm conditions existed at the inland location. In contrast, smoke from a nearby inland fire was rising vertically and at a height of about 100 feet above the ground was carried vigorously inland by the upper onshore flow. In our opinion, the inland exposure measures an extremely localized condition and cannot be used to estimate either diffusion or transport parameters at the shoreline or at the inland site boundary beyond the dunes.

In general, a site on the shore of a large body of water such as the Great Lakes is unique for several reasons. First, for the purpose of safety evaluation, the onshore wind directions are of primary concern. Second, in comparison to offshore flow the long fetch of smooth water surface will tend to increase the onshore wind speeds at the shoreline followed by a decrease in the winds inland because of increased surface friction. Third, onshore flow is generally expected to be less turbulent than offshore flow but as the flow moves inland turbulence is added in the lower few hundred feet because of surface roughness. Thus one would expect the average annual onshore wind speed of 16.4 mph as measured at the shoreline location to be about 25% less at the inland site boundary. The inland measuring site discussed above recorded an average annual wind speed of 10.2 mph but because of exposure we consider the location to be non-representative of inland transport.

The combination of onshore and inversion conditions occurs 20% of the time according to the shoreline measurements. If this combination is restricted further to winds below 4 mph the frequency is about 1%. Thus one can conclude that an atmospheric diffusion rate equal to or less than that equivalent to moderate inversion conditions (Pasquill Type F) and a 2 m/sec wind speed occurs about 1% of the time if limited to onshore flow. The resulting relative ground air concentration at the nearest site boundary (700 m) is 6×10^{-4} sec m^{-3}. Using the applicant's diffusion parameters for the first 2 hours (page 14.22-3), the resulting site boundary concentration would be 3×10^{-4} sec m^{-3}, a factor of 2 less than our computation. Neither computation accounted for the effect of the building wake. Since the applicant's expression for the virtual source distance (middle of page 14.22-3) is in error - dimensionally incorrect and obviously gives an insignificant number - we are unable to asses the dilution factor credited to the building wake effect that was used. Using a shape factor of 1/2 and containment building cross-sectional area of 2210 m, our assessment of the added wake dilution is a factor of 2.3 at the site boundary under inversion conditions.

For the period from 2 to 24 hours, the meteorological statistics from the site show that wind speeds less than 2 m/sec persisted for 19 hours on one occasion during the year. Wind direction persistence for the region [1] shows that on the basis of 5 years of data, the maximum time that the wind remained in a 22-1/2° sector was a single occasion of about 30 hours. However, the more persistent winds, directionally speaking, tend to occur with higher wind speeds. Thus, in our view, it would be reasonably conservative to assume that for the 2 to 24 hour period at this site the concentration would be averaged over a 22-1/2° sector with inversion conditions (Pasquill Type F) and a 4 m/sec wind. These parameters are approximately equivalent to the applicant's Sutton parameters listed on page 14.22-4 if the reflection term is used in Sutton's diffusion equation.

In summary, we have had difficulty in assessing the diffusion rates used by the applicant to compute downwind radioactive doses because 1) no listing of relative concentration (χ/Q) was given, 2) the diffusion equations listed in Section 14 and Appendix D do not agree and, 3) the virtual source distance expression appears to be in error. Our computation of the site boundary relative concentration for the first 2 hour period is 2.6×10^{-4} sec m^{-3} taking into account building wake effect. For the 2 to 24 hour period the value was computed to be 6×10^{-5} sec m^{-3} on the basis that it is highly improbable that the triple restriction of low wind speeds, inversion conditions and wind direction constant in a 22-1/2° sector could occur simultaneously over a 24-hour period.

Reference

[1] Van der Hoven, I. (1969), "Wind Persistence Probability". To be published as ESSA Research Laboratories Technical Memorandum.

Comments on

Palisades Plant
Consumers Power Company
Final Safety Analysis Report
Amendment No. 12 dated May 14, 1969

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
June 13, 1969

The additional meteorological data presented in Amendment 12 consists of a two-month (September 1967 and February 1968) analysis of the wind direction standard deviation as derived from the wind direction range. We find no experimental evidence for the applicant's assumption of a 4.3 value for the range to standard deviation ratio (R/σ_θ). Markee [1] found from an analysis of 1-to 15-sec wind direction readings at a number of locations that the average R/σ_θ value ranged from 5.0 to 8.6. This, apparently, is the basis for the commonly-used value of 6.0 which would decrease the applicant's σ_θ values by a factor of 0.7. In the case of Table 11, where four categories of $\sigma_\theta\ \bar{u}$ are shown assuming a $\bar{u}$ of 4.47 mph, the upper limits of the σ_θ value of the first three categories would then be 1.8°, 2.7°, and 3.6°, respectively, and only the last category could be classified as a diffusion rate greater than Pasquill Type F. Thus, for the two months shown in Table 11, onshore winds with diffusion rates equivalent to Type F and 2 m/sec or less occur 6.6 percent of the time.

As pointed out in our comments of February 3, 1969, we consider the inland anemometer site to be non-representative of inland transport because of the extreme sheltering effect of the sand dunes and dense vegetation. To what extent the $\sigma_\theta\ \bar{u}$ values are affected by the opposite factors of decreased wind speed and increased wind variability at the inland site boundary is not known. Doubtless, wind speeds will decrease and turbulent fluctuations of wind direction will increase as the flow over a relatively smooth water surface becomes heated over the rougher land surface. It is interesting to note that for onshore winds and stable conditions as represented by the first three categories in table 12, the inland station shows a frequency of 9.5 percent. This compares to the previously mentioned 6.6 percent for the shoreline station.

In summary, we have not seen any evidence to change our conclusions as expressed in the February comments. Our computation at the site boundary (700 m) for the first two-hour period is a relative concentration of 2.6×10^{-4} sec m^{-3}, assuming Type F diffusion a 2 m/sec wind speed, a building shape factor of $\frac{1}{2}$ and a cross-sectional area of 2210 m^2. We do not feel that the inland station as presently located is representative of the air flow at the inland site boundary.

Reference

[1] Markee, E. H., 1963: "On the relationships of range to standard deviation of wind fluctuations". Mon. Wea. Review, 91(2), pp. 83-87.