237. State the name of each consulting firm, individual and agency who was requested by you to analyze the proposed Midland Units. For each such consulting firm, individual and agency state what area or problem of the proposed Midland Units it analyzed and what the results were. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

The following agencies and consulting firms were requested to analyze the proposed Midland Units:

1. Environmental Science Service Administration (ESSA)

2. U. S. Geological Survey (USGS)

3. U. S. Coast and Geodetic Survey (USC&GS)

4. U. S. Fish and Wildlife Service

5. John A. Blume and Associates, Engineers

The final reports of the agencies and the consulting firms are attached to the Safety Evaluation as Appendices C through G. Other reports, preliminary in nature, are among the documents covered by Question II of the Atomic Safety and Licensing Board's certification of questions to the Atomic Safety and Licensing Appeal Board, dated June 22, 1971.

Joseph A. Murphy

Joseph A. Murphy

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240. With respect to the statement "The consequences of these transients will be calculated again when detailed plant design information is available to verify that these transients are within the capabilities of the reactor control and protection systems," at page 59 of the Safety Evaluation, state each fact and assumption which supports your belief that a complete analysis of the final design insofar as transient stability is concerned, is not an important safety factor to be considered completely prior to any recommendation approving the proposed Midland Units. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

As stated on page 59 of the Safety Evaluation the following plant operating transients have been analyzed by the applicant:

1. Uncompensated reactivity changes resulting from fuel depletion and changes in fission product poison concentrations (Ref. PSAR, Section 14.1.2.1),
2. Control rod withdrawal during startup and at power (Ref. PSAR, Sections 14.1.2.2, 14.1.2.3, Answer 13.1.1, 13.1.2),
3. Dilution of the boron concentration in the coolant (Ref. PSAR, Section 14.1.2.4, Answer 13.1.3).
4. Startup of an inactive coolant loop (Ref. PSAR, Section 14.1.2.5),
5. Loss of Coolant Flow (Ref. PSAR, Section 14.1.2.6),
6. Malpositioning of a control rod (Ref. PSAR, Section 14.1.2.7),
7. Loss of ac electric Power (Ref. PSAR, Section 14.1.2.8),
8. Loss of electrical load (Ref. PSAR, Section 14.1.2.8).

These analyses indicate that no fuel damage will result from these operating transients. Each is preliminary in that the analyses were based upon design parameters of the facility which were available at the time the analysis was performed. Considering that the analytical procedures for analysis of these transients are well established and that the design parameters are not expected to change significantly, we do not expect the results of the analyses of these transients when based upon final design parameters to be significantly different from those analyzed on the basis of preliminary design information. However, a final evaluation to assess the risk to public health and safety, taking into account any pertinent information developed since the submittal of the PSAR, will be performed at the time of our operating license review to verify that no fuel failure will result.

Joseph A. Murphy

242. With respect to the statement at page 60 of the Safety Evaluation that "the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations," state:

(a) What is lowest level of possible accidental dose which is contemplated by controlling the limiting of activity as aforesaid;

(b) What level of activity concentration will achieve the dose set forth in (a) above; and

(c) Can such levels be lower than set forth in (b) and if so state why, if it is true, you do not intend to seek to impose such lower levels.

State each fact, calculation and assumption upon which you base your answer. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

(a) Concentrations would be limited to that which, in the unlikely event of coolant release, would not result in individual doses offsite in excess of 0.5 rem external dose to the whole body or 1.5 rem inhalation dose to the thyroid.

(b) The primary coolant would have concentration limits of $22/\bar{E}$ uCi/cc for total activity and 0.017 uCi/cc for iodine - 131. (See Attachment 1). The secondary coolant concentrations would be adequately limited by the primary coolant limits.

(c) Although lower limits can be achieved, we consider the above limits to be sufficiently low, in view of the low probability for the occurrence of the accidents under the assumed conditions.

The calculated limits represent concentration levels less than those estimated by the applicant for 1% failed fuel.

A. P. Kenneke

ATTACHMENT 1

Primary Coolant Limits to Limit Consequences of Steam Generator Tube Rupture

A. Total Activity Limit

$$C_p = \frac{2 \cdot D \cdot P}{\bar{E} \cdot f \cdot V_p \cdot \chi/Q \cdot a_1 \cdot a_2 \cdot a_3}$$

Where C_p = primary coolant concentration in uCi/cc.

2 = dosimetric geometry factor

D = whole body dose criterion = 0.5 rem

P = density of air in gm/m^3

$\bar{E}$ = average energy of radiations from the nuclides in the coolant in Mev/dis

f = fraction of coolant assumed lost through the steam generator tube rupture before it is isolated = 0.5

V_p = total primary coolant volume = 334 m^3

χ/Q = meteorological diffusion factor between point of release and the site boundary = 6.7×10^{-4} sec/m^3

a_1 = physical constant = 3.7×10^{10} dis/sec/Ci

a_2 = conversion factor = 1.6×10^{-6} erg/Mev

a_3 = conversion factor = 10^{-2} gm-rem/erg

B. Iodine Activity Limit

$$C_p = \frac{D}{f \cdot V_p \cdot B \cdot \chi/Q \cdot DCF \cdot 1.5 \cdot p}$$

Where

C_p = Iodine 131 concentration in primary coolant in uCi/cc

D = thyroid dose criterion = 1.5 rem

f, V_p, χ/Q = defined as above for whole body calculation

B = standard breathing rate 3.47×10^{-4} in m^3/sec

DCF = dose conversion factor for I-131, 1.48×10^6 in rem /Ci

1.5 = factor to account for the contributions of I-132 - 135

p = fraction of iodines in coolant assumed to escape to the atmosphere = 1

247. With respect to your analysis of the Dow emergency plan, which you refer to at page 70 of the Staff Safety Evaluation, describe in detail each fact, calculation and assumption by which you conclude that the dose that might be received by an employee standing one mile from the reactor during 35-minute and one-hour periods following a design basis LOCA would be, respectively, 55 rem to the thyroid and 75 rem to the thyroid. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy. [to the extent that information is sought as to the basis of the calculation of the dose].*

The dose to an individual at one mile from the reactor is calculated as follows:

The dose at 2 hours at the exclusion distance (500 m) was 270 rem, based on a X/Q for Pasquill Type F stability and a 1 m/s wind speed of 6.7×10^{-4} and a spray reduction factor of 3.1.

At one mile, the X/Q for Pasquill Type F stability and a 1 m/s wind speed is 2.3×10^{-4}. The spray reduction factor for one hour is 2.2 and for 35 minutes is 1.7. Thus, the doses are:

a. For 35 minutes:

$$270 \times \frac{2.3}{6.7} \times \frac{35}{120} \times \frac{3.1}{1.7} = 50$$

b. For 1 hour:

$$270 \times \frac{2.3}{6.7} \times \frac{1}{2} \times \frac{3.1}{2.2} = 65$$

* The 35 minute and one hour dose calculations have been modified to 50 rem and 65 rem, respectively. See Tr. 1675.

A. P. Kenneke

255. Describe in detail each "Improved means for prompt detection of fuel clad failure" which you say in the Staff Evaluation is under development within the industry. What percentage of leaking fuel rods can the presently considered process radiation monitor detect? What increase in coolant activity, as the system is presently designed, can occur without being detected. If in your answer you make reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

Industry is investigating the following types of failed fuel element detection systems:

1. A delayed neutron monitor based on detection of delayed neutrons emitted by several fission products following beta decay.
2. A differential-gamma monitor which separates fission product gamma radiation from normal activation product gamma radiation on the basis of energy.
3. An off-gas system monitor based on the beta detection of gaseous fission products stripped from a sample stream.
4. An ion-exchange gamma monitor based on sorption of fission products on ion exchange resins and gamma energy discrimination.
5. A Cerenkov monitor system based on detection of high energy beta particles from selected fission products.
6. A gamma monitor for detection of gross gamma activity and of selected isotopic activity by gamma energy discrimination.
7. A gross gamma monitor without energy discrimination.

Developmental efforts are proceeding primarily on items 1, 6 and 7 listed above.

As indicated on page 7-34 of the PSAR, a study is being conducted by Babcock & Wilcox to determine the source strengths of the various isotopes released upon fuel failure to permit determination of the required sensitivity of the detector. As indicated on page 19 of the Safety Evaluation, we will require that the applicant provide a system having detection sensitivity equivalent to that of the best equipment available to detect promptly the gross failure of a fuel element at the time of the operating license review.

Since the sensitivity of the monitor has not yet been chosen, we cannot respond to the inquiries on the detectable fraction of leaking fuel or the increase in coolant activity which could occur without detection.

REFERENCES:

(1) A Process Radiation Monitoring System for Large PWR's, C. H. Meijer and B. Hekkala, Paper presented at the American Nuclear Society Power Reactor Systems and Components Meeting, Williamsburg, Virginia, September 1-3, 1970 (copy attached).

(2) WCAP-7614-L, Topical Report-Safety Related Research and Development for Westinghouse Pressurized Water Reactors - Program Summaries - Fall 1970 (PROPRIETARY)

Joseph A. Murphy

Joseph A. Murphy

265. What are each of the "uncertainties" in the calculated peak in the containment structure during a LOCA as you so state at page 23 of the Staff Safety Evaluation. Also state what steps you and Applicant are taking or proposing to take to resolve each such uncertainty and what relationship, if any, the non-resolution of each such uncertainty has to the safety of the proposed Midland Units. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

Items which cause possible uncertainties in the calculation of peak containment pressure are of two general types: (1) those resulting from the use of preliminary rather than final design information, e.g., containment free volume and the surface area available for heat transfer, and (2) those resulting from limitations of the physical and mathematical models used in the analysis. Uncertainties resulting from the former will be eliminated when detailed design information is available. With the respect to the latter, in accordance with Criterion 50 of the General Design Criteria, Appendix A to 10 CFR 50, we require the containment to be designed with sufficient margin to reflect

(1) the effects of potential energy sources which have not been included in the determination of peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning,

(2) the limited experience and experimental data available for defining accident phenomena and containment responses, and

(3) the conservatism of the calculational model and input parameters.

Joseph A. Murphy

284. Describe in detail your evaluation and analysis, stating each fact, calculation, and assumption thereof, of the probability and consequences of "these types of events" as stated at page 45 of the Safety Evaluation which will provide the basis for further review of the proposed design of the systems regarding their ability to terminate or limit the consequences of such events. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

We presume the Interrogatory refers to the probability and consequences of "these types of events" as stated on page 48 of the Safety Evaluation. The phrase "these type of events" does not appear on page 45.

Our evaluation and analysis of the probability and consequences of failure to scram in the event of anticipated transients has not yet been performed. We have identified the additional information we require to permit us to perform this evaluation and have requested this information from the vendor of the nuclear steam supply system, The Babcock & Wilcox Company, in the attached letter dated December 14, 1970. As noted in the Safety Evaluation, our evaluation, when complete, will provide the basis for further review of the design of plant systems regarding their ability to terminate or limit the consequences of failure to scram in the event of anticipated transients. We will require the applicant to make such changes in the final design, if any, as may be found necessary as a result of this further review.

Joseph A. Murphy

Joseph A. Murphy

DEC 14 1970

Mr. John H. MacMillan
Babcock & Wilcox Company
P. O. Box 1260
5061 Fort Avenue
Lynchburg, Virginia 24505

Dear Mr. MacMillan:

During the past two years, each of the nuclear steam supply system designers has performed some investigations of the subject of anticipated transients without scram (ATWS). These investigations were performed in response to concerns that have been raised regarding the possibility of such occurrences and the capability of nuclear facilities to cope with them if they were to occur. The several investigations completed to date have not been performed on a consistent basis.

We need additional information, developed on a consistent basis, to aid in our further assessment of this problem and in reaching a conclusion as to whether protection against ATWS need be considered as a design basis requirement for nuclear facilities. We request that you provide the results of an investigation of ATWS conducted in accordance with the guidelines listed in the enclosure to this letter. With minor exceptions, the guidelines listed are identical to those discussed with your representatives earlier this year.

We consider the need for the results of the requested investigations to be of relatively high priority. In view of your prior work on this subject, we believe that it can be completed in a timely manner. If you desire to discuss this matter further, please contact R. C. DeYoung, Assistant Director for Pressurized Water Reactors, Division of Reactor Licensing.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

DEC 14 1970

GUIDELINES FOR INVESTIGATION OF ATWS

(1) The study should include the following postulated occurrences:

(a) Loss of load (turbine trip along with or caused by condenser loss, i.e., no steam dump).

(b) Load increase (opening of largest secondary valve, e.g., bypass valve).

(c) Loss of feedwater

(i) failure of 1 pump or feedwater system valve

(ii) failure of all pumps or feedwater system valves

(d) Loss of primary flow (coast down)

(i) loss of 1 pump

(ii) loss of all pumps

(e) Total loss of power from offsite sources (assume simultaneous loss of power from the nuclear unit, and that power is available from the diesels, but that control rods do not move regardless of loss of power).

(f) Inactive loop startup - unless precluded by interlocks designed to IEEE-279 protection system criteria.

(g) Rod withdrawal at zero power.

(h) Rod withdrawal at full power.

(i) Opening of the largest single valve in the primary system a combination of valves that could open as the result of a single fault (depressurization).

(j) Control rod maloperation (dropped rod, stuck rod, operator error in use of part-length rod).

(k) Boron dilution.

(l) Loss-of-coolant resulting from a break in a small pipe (largest instrument or sampling line).

(m) Other transients of the same order of probability.

(2) Each category of occurrence should be examined over the full range of possible initial power and flow conditions and over the range of possible magnitude and rate of the initiating occurrence. Sufficient information should be presented to indicate the conditions within this full range for which additional measures are required to preclude or reduce the consequences of the occurrence. If such measures are proposed, it should be demonstrated that they can be effective over the required range.

(3) As an aid to understanding the probability of each occurrence, a discussion should be provided of the sequence of events and failures that must be experienced for each postulated transient as well as a justification for the limits selected for the occurrence (e.g., upper limits used for number of rods withdrawn either simultaneously or sequentially in the reactivity transients).

(4) The assumption should be made that all other systems, including control systems, react normally unless prohibited as a normal consequence of the transient. (However, no rods should be assumed to move inward at any time either by scram or drive action.) The study should define the systems that are assumed to function (and not to function) at all times during the course of the transient. It would be helpful if the analyses would clearly identify the systems that are assumed to function in limiting the consequences of the event and the magnitude of the beneficial effect produced by the operation of each such system.

(5) Assumed initial conditions and system parameters (e.g., power level, flow, pressure, power distribution, feedback coefficients) need be no more stringent than those normally anticipated for the reactor state under consideration.

(6) The course of each transient should be evaluated assuming operation of all available systems until the transient has been shown to have been terminated successfully (defined as essentially zero power with the reactor in a coolable geometry with normal cooling in operation and the containment pressure within design limits) either with the usually available systems or with system(s) especially devised for use with this class of occurrence, or until it has been shown that the ultimate conclusion of the transient would result in an accident whose radiological consequences would be worse than the guidelines of 10 CFR Part 100. For acceptable conclusions it should be

justified that all reactor system state conditions (e.g., fuel temperatures, pressures) reached during the transients that exceed normal design limits do not jeopardize the integrity of the system to assure that the transient does not become compounded (e.g., pressures are not high enough to increase significantly the probability of a loss-of-coolant accident). Where acceptable conclusions are not reached, it would be useful to explore the feasibility of mitigating the consequences of the transient by additional design features.

289. With respect to the Applicant's gaseous release rates as set forth at page 52 of the Safety Evaluation, what will be the maximum concentration at anytime (and not averaged over any period) of radionuclides at the site boundary. Describe in detail each fact, calculation and assumption upon which you base your answer. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

The average release rate for Xenon 133 given on page 52 of the Staff Safety Evaluation is estimated by the applicant to result in an annual average concentration at the site boundary of 2.5×10^{-7} microcuries/cc. Since meteorological conditions are more or less randomly distributed in time, only a probabilistic statement about the instantaneous concentration at the site boundary can be made. For a continuous source, turbulent eddies of practically all sizes can affect the diffusing plume (e.g., in emission over infinite time with an infinitely long time of sampling,all possible eddy sizes would be effective). However, for the realistic case of a continuous plume with a finite sampling time considerably less than the plume release time, the fraction of the turbulence "seen" at the sampler is a strong function of sampling time.

As stated in the safety evaluation on page 53, on-site meteorologic data are not yet available. However, assuming average meteorological data for the site and that the facility did operate at the release rate of 42.5 millicuries per second of Xenon 133, we calculate that instantaneous concentrations at a point <u>chosen at random</u> on the site boundary could exceed 2.5×10^{-5} uCi/cc (100 times the annual average concentration) 5% of the time.[1/] The probability that any particular point would experience a concentration in excess of 2.5×10^{-5} uCi/cc is approximately one chance in 50.[2/] The concentrations of Krypton 85 and Krypton 88 would be proportionally lower than the Xenon 133 concentration. Ultimately, dose is the important consideration, and concentration, averaged over both time and space, determines the dose received. The longer the averaging period, the greater the probability that the wind direction will vary away from the selected point, and the concentration will approach the annual average value at the point of highest concentration on the boundary.

1/

The maximum point concentration at anytime at a distance R, from a continuous source, Q, is given by the following equation:

$$X = \frac{Q}{\pi \mu \sigma y \ \sigma z}$$

Where X = Concentration (Ci/m^3)
Q = Release rate (Ci/sec)
σ_y = Horizontal standard deviation of Gaussian plume (m)
σ_z = Vertical standard deviation of Gaussian plume (m)
μ = Wind speed (m/sec)

The values of σ_y and σ_z are functions of the atmospheric stability and the distance from the source. The above estimate is based on having Type F stability and a 1 meter per second speed wind speed.

2/ The plume width can be taken approximately as $2\sigma_y$; the site circumference is $2\pi R$. The probability is then $\sigma_y/\pi R$. For Type F at 500 meters, this would be 20/3.2 x 500 = 20/1600, or less than 1/50.

The above is based on an assumed release rate estimated by the applicant. At the time of our operating license review, we will require the applicant to comply with all requirements of 10 CFR Parts 20 and 50 regarding effluent releases then in effect.

A. P. Kenneke

A. P. Kenneke

303. Explain in detail your statement, "Babcock & Wilcox Company will have day to day responsibility for the nuclear steam supply system," which appears at page 72 of the Safety Evaluation. Include within your answer how such "day to day responsibility" affects applicant's overall responsibility for the design and construction of the proposed Midland Units including quality assurance responsibility as as set forth in Appendix B to Part 50 of AEC regulations. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

The Babcock & Wilcox Company will perform the basic quality assurance program for the Nuclear Steam Supply System. This program covers the independent audit of quality control programs during design, procurement, fabrication, and testing. Final responsibility for the quality assurance program rests with the applicant, which will exercise this responsibility by reviewing each principal contractor's quality assurance program and by performing audits and surveillance of the contractors to assure proper implementation of the program. In addition, Bechtel Corporation on behalf of the applicant will audit the quality assurance efforts of the Babcock & Wilcox Company.

As stated on page 73 of the Safety Evaluation, we conclude that the Midland plant quality assurance program meets the requirements of the "Nuclear Power Plant Quality Assurance Criteria," Appendix B, 10 CFR 50 and is acceptable.

Joseph A. Murphy

Joseph A. Murphy

304. Explain in detail your statement, "B & W will also audit the quality assurance programs of its suppliers as appropriate." which appears at page 73 of the DRL Safety Evaluation. Define "appropriate" as it is used, including each standard and criteria. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

/except insofar as it asks for "each standard and criteria"/

As indicated on page 1B-12d and 1B-12e of the PSAR, Babcock & Wilcox performs pre-awards audit and post-award inservice auditing and product surveillance of supplier manufacturing programs to assure compliance with written procedures previously approved. The degree and frequency of surveillance and auditing is determined by the safety significance of the component and the past performance of the supplier.

Joseph A. Murphy

Joseph A. Murphy

325. What limits are or will be imposed upon Dow expansion plans beyond which would require a review or evaluation of such expansion upon the safety of the proposed Midland Units. If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, or reference to other than textual (exclusive of footnote) matter in your Safety Evaluation, then set forth completely the text of each such reference or attach a copy.

A small portion of the exclusion area falls within the Dow Chemical Company property. Consumers Power Company will exercise the right to remove any persons from this Dow property when conditions arise which warrant removal of persons from within the exclusion area. We will require the applicant to inform us and to evaluate the situation if any change in the status of that portion of the exclusion area involving Dow Chemical Company property occurs. In other respects, no specific restrictions will be placed upon the Dow Chemical Company. However, the applicant will be required to review and evaluate and inform the Commission of any modification at the Dow Chemical plant which could create an unreviewed safety question. Thus, if any change at Dow affects the ability to evacuate the exclusion area, or creates an additional explosive, toxic, or other hazard at the Midland plant, the applicant will be required to review and evaluate the activity and if necessary modify the plant or its operation to assure that potential accidents will not affect the nuclear plant.

Joseph A. Murphy

Joseph A. Murphy