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23. State each fact, calculation and assumption, including the criterion and design philosophy or design basis upon which you conclude that "a system to mitigate the consequences of a vessel failure due to thermal shock following a Loss of Coolant Accident is not justified." (PSAR p.4-15) In addition:

(a) If your answer is based in whole or in part upon historical precedent, identify:

(1) Each document which refers or relates to or demonstrates this precedent;

(2) Each oral communication which refers or relates to or demonstrates this precedent and give, regarding each such communication, the date and place thereof, the identity (by name, address, by whom employed, with what group or organization affiliated and for whom acting) of each person involved therein, and the complete substance of what was said by and to each person.

(b) If your answer is based in whole or in part upon financial, economic or engineering factors, identify:

(1) Each document which refers or relates to or demonstrates each such factor:

(2) Each oral communication which refers or relates to or which demonstrates each such factor and give, regarding each such communication, the date and place thereof, the identity (by name, address, by whom employed, with what group or organization affiliated and for whom acting) of each person involved therein, and the complete substance of what was said by and to each person.

If in your answer you make reference to other than textual (exclusive of footnote) matter in the PSAR, then set forth completely the text of each such other reference or attach a copy.

We have evaluated the Babcock & Wilcox Topical Report BAW-10018, "Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock." As a result, we have concluded at this time that there is reasonable assurance that thermal shock will not result in vessel failure and that installation of equipment to accommodate such a failure is not required. Continuing work, however, is being performed under the Heavy Section Steel Technology (HSST) program by Oak Ridge National Laboratory on the fracture toughness properties of irradiated steel. Although not anticipated, the results of this work could possibly cast doubt on the ability of the irradiated reactor vessel to maintain its integrity. In view of this possibility the applicant (1) has reviewed the design of the plant to determine that it will be feasible to anneal the vessel so as to reduce the deleterious effects of irradiation and (2) will include provisions in the design of systems and structures to permit the future addition of a system to assure continued core cooling in the event of a reactor vessel failure by thermal shock from emergency core cooling water.

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241. With respect to your statement at page 60 of the Staff Safety Evaluation that "Based on our evaluation of the information submitted by the Applicant and our evaluations of other pressurized water reactor designs at the operating license stage," describe in detail the evaluations of these other pressurized water reactor designs insofar as you contend such evaluations relate to the proposed Midland Units. Include within your answer the name of each pressurized water reactor you have relied upon and whether you have relied upon anything in Compliance Division inspection Reports regarding such other reactors and if you have, then list the dates of such inspection reports. 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 such reference or attach a copy.

Our review and evaluation of the recently licensed Ginna, H. B. Robinson Unit 2, Palisades and Point Beach Unit 1 plants confirm that the protection and control systems for these plants, as finally designed, are such that the operating transients referenced on page 59 of the Safety Evaluation and listed in response to Question 240 can be terminated without the core and reactor coolant boundary being damaged, and with no significant off-site radiological consequences. The operating license review for the foregoing plants was being conducted concurrent with our review of the Midland Plant; however, none of these plants were designed by B&W. Subsequent to completion of the Safety Evaluation for the Midland Plant, we reviewed the Duke Power Company Oconee Unit No. 1 FSAR, which is a B&W design, with regard to the operating transients and again confirmed our previous conclusion regarding the adequacy of the protection and control systems.

318. Identify and list each standard, objective and criteria pursuant to which you evaluate the PSAR and the design and proposed construction of the proposed Midland Units. State whether any such standard, objective and criteria is different from those used in connection with evaluation or review or approval of any license or permit for each other PWR plant which you rely upon in any way in connection with the Midland Safety Evaluation. 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 special considerations, if any, given to the Midland Plant]

As indicated in the Safety Evaluation, many aspects of the preliminary design of the Midland Plant are similar to those of plants previously reviewed and approved by the Regulatory Staff. We have based our review of these areas on our earlier review of similar areas in other applications. The standards, objectives, and criteria used in the evaluation of the other plants are the same as those employed for our review of the Midland Plant with three exceptions:

(1) Site-related items - Because of the population density close to the Midland reactor, we required the installation of the following engineered safety features, some of which have been required for PWR plants which are situated in relatively populated areas.

A. Addition of chemical additives to the containment spray system.

B. A penetration pressurization system.

C. An isolation valve seal water system.

D. Charcoal filters for fuel storage pool ventilation exhaust.

E. Pressurized weld channels over the seam welds between the containment liner plates.

F. Design provisions for the installation of a post-loss-of-coolant accident reactor vessel cavity flooding system.

In addition, we required a reduction in the design leakage rate of the containment and reduction in the minimum exclusion distance and in the distance to the outer boundary of the low population zone.

(2) Chemical releases - Because of the proximity of the Dow Chemical Plant to the Midland reactors, special consideration was given to the protection of control room personnel from the potential consequences of accidents at the chemical plant. The standard of acceptability of the concentrations of toxic chemicals in the control room following a release at Dow are the

Threshold Limit Values established by the American Conference of Governmental Industrial Hygienists.

(3) Codes - The reactor coolant piping of the Midland Plant will be designed to the ANSI B31.7 Nuclear Power Piping Code. Some earlier facilities were designed to the Power Piping Section of the USA Standard Code for Pressure Piping, B.31.1. ANSI B31.7, the more recent of the two codes, was developed specifically for nuclear power applications.

The applicant has stated that in-service inspection will be conducted in accordance with the "Draft ASME Code for In-Service Inspection of Nuclear Reactor Coolant System (N-45). This draft code is equivalent to Section XI of the ASME Boiler and Pressure Vessel Code. We will require that the detailed in-service inspection plans comply with Section XI.

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