December 15, 1972

NOTE TO R. DEYOUNG

RESPONSE TO INTERVENER'S CONTENTIONS

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER AND LIGHT COMPANY
MADISON GAS AND ELECTRIC COMPANY
DOCKET NO. 50-305, KEWAUNEE NUCLEAR POWER PLANT

The Mechanical Engineering, and Materials Engineering Branches has prepared the enclosed responses to the contentions #3,4.1-K, 3.4.2, 3.4.3, 3.3.10, 3.3.11, 3.3.12, stated in the intervener's (Business for the Public Interest) letter of December 11, 1972. These responses have been prepared within the limited time available, in accordance with the urgent request received from L. Crocker, reactor project manager for Kewaunee Nuclear Power Plant.

Original signed by
R. R. Maccary

R. R. Maccary, Assistant Director
for Engineering
Directorate of Licensing

Enclosure:
Responses to the Contention's

cc: w/encl.
J. Hendrie
S. Hanauer
K. Kniel
L. Crocker
S. Pawlicki
D. Lange
M. Fairtile

DISTRIBUTION
L Subject File
L Reading File
MTEB R/F
MEB/R/F
PDR Room

OFFICE ▶	L:MEB	L:MTEB	L:AD/E			Memo
SURNAME ▶	DLange:amp	SPawlicki	RRMaccary			
DATE ▶	12/15/72	12/15/72	12/15/72			

Response to Contention 3.3.10

In the Point Beach Internal Assurance Program Westinghouse attempted to utilize the nuclear noise method and the accelerometer measurements on the vessel for loose part monitoring. Results indicate that the nuclear noise method is not applicable. The signal can only indicate the gross effect of core vibration but interpretative distinction of the significant response modes and detailed identification of loose parts cannot be achieved. In the case of accelerometers, the signals can not be discriminated from the background noise. In fact, due to the extreme thickness and stiffness of vessel wall, any signals obtained from externally mounted sensors will not detect the vibration response of the internals, and are representative only of the rigid modes of response of the vessel on its supports.

The inspection program following preoperational testing as outlined in response to contention 3.3.11 includes inspection for loose parts after exposing the internals to the endurance cyclic limit. Fatigue failure of the internals will not occur after exposure to this cyclic limit since the component parts should be adequate to withstand an infinite number of additional cycles.

In addition, periodic internal inspection of reactor internal during each refueling provides a means for examination of the interior spaces of the reactor for further verification that loose parts are not present.

Response to Contention 3.3.11

AEC Safety Guide 20 provides guidance to confirm that the reactor internals are designed to withstand the predicted cyclic loads due to vibration encountered during their service life. A preoperational test is required with sufficient instrumentation to confirm that the vibration characteristics of the Kewaunee internal structures are similar to those of the Ginna and the standard 2-loop plant. In addition, the preoperational testing program includes an inspection program to insure that structural degradation had not occurred after the internals have been subjected to more than 10^7 cycles of loading. Since this cyclic loading represents an associated stress amplitude below the endurance stress amplitude limit the internals may be expected to withstand an infinite number of additional application of these loadings without failure in service.

In addition, a structural dynamic analysis using the LOCA forcing functions derived from hydrodynamic analysis which have been confirmed by testing has been completed on the Kewaunee plant. The associated stress levels were well within the limits specified by the ASME Codes. Therefore, vibration induced damage is not expected to occur from forcing functions introduced during normal, upset, emergency and faulted conditions on the reactor internals. The forcing functions may be derived from measured responses and analytical and empirical results. Successful utilization of (a) past experience of full scale and reduced model testing of different plant designs, (b) the preoperational test program required for Kewaunee, (c) instrumentation requirements to confirm that Kewaunee has similar response characteristics to Ginna,

(d) the satisfactory reactor internals operation of Ginna plant and

(e) the inspection program of the preoperational testing program will verify the structural integrity and fatigue design of the Kewaunee components of the reactor coolant pressure boundary.

The internals and all the reactor coolant boundary components (vessels, pumps, steam generators, etc.) have been designed to withstand a combination of normal operation plus design basis accident plus design basis earthquake loadings within the limits of the ASME Code. In consequence, the use of accelerometers to measure the extremely low strains present in components during normal operation would have little significance. Additional accelerometers located on the steam generators, reactor coolant pumps and coolant piping as you suggested will also not be helpful for monitoring the vibration behavior of core internals because of their remote location from the reactor internals. Further detailed discussions are given in our response to Contention 3.3.10.

Fatigue failures of reactor internals that have occurred at several plants were due to inadequate thermal shield and guide tube designs. The Kewaunee thermal shield is a one piece rigid attachment and support design which have fundamental frequencies far removal from the flow induced forcing frequencies. In addition to the above testing and analytical requirements, the internals are equipped with a secondary core support device that is designed to prohibit core dropping that would inhibit cooling.

Response to Contentions 3.4.1-K and 3.4.2

The Staff recognizes that measures taken to assure the integrity of pressure vessels in light water reactor plants must be properly considered in licensing proceedings. The purpose of such consideration is to determine whether the measures taken are sufficient to demonstrate that the pressure vessels can be operated over their service lifetimes with a negligible risk of failure.[1/]

Specific aspects of nuclear faility design which must be considered in assessing the acceptability of facility design for the purposes of Commission licensing are set forth in the General Design Criteria, Appendix A to 10 CFR Part 50. No provision of the General Design Criteria requires protection against the consequences of a failure of a reactor vessel. Rather, a number of the key provisions of the Criteria require designs of such character as to minimize the likelihood of such a failure. (Refer particularly to Criteria 14, 15, 30, 31, and 32.)

1/"Failure" is defined by the Staff as pressure vessel rupture of such an extent that the capability of emergency core cooling systems adequately to cool the core may be impaired.

Evaluation of the application included assessment that the facility characteristics were in conformity with Parts 50 and 100. Such assessment led to the determination that the design characteristics of the facility were such that consequences of accidents assumed for purposes of determining siting suitability are not exceeded by those of any accident considered credible. Careful consideration was given to the particular aspects and features of the specific facility and site to assure that all credible events were considered.

Under the Commission's Regulations there is no requirement for protection against the consequences of pressure vessel failure unless it has been determined that there are special considerations that make it necessary that potential pressure vessel failure be considered.

The Staff, in its evaluation, reviewed all factors which contributed to the structural integrity of the reactor vessel concludes that because of conformance with the rules of ASME Section III construction rules for nuclear vessels, no other special consideration exists for which failure of the reactor pressure vessel need be considered in the analysis of a loss-of-coolant accident for Kewaunee Nuclear Power Plant.

Response to Contention 3.4.3

In all cases evaluated to date, the Staff has concluded that the probability of failure of the pressure vessel is so low as to be incredible and has not required consideration of the consequences of such failure in the assumptions employed in determining site suitability.

The basis for the Staff's confidence concerning the low probability of nuclear pressure vessel failure includes the fact that design, material, fabrication, inspection, and quality assurance requirements for nuclear pressure vessels are specified in Section 50.55a of the Commission's Regulations, in Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants," in Appendix B to Part 50, "Quality Assurance Criteria for Nuclear Power Plants," and the operating limitations are determined in accordance with the proposed Part 50, Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements." The integrity of the Kewaunee pressure vessel is assured, since this reactor vessel:

1. Was designed and fabricated to the high standards of quality required by the 1968 Edition of Section III of the ASME Code, the 1968 Winter Addenda and Code Cases and the Westinghouse equipment specification.

2. Was made from materials of controlled and demonstrated high quality.

3. Underwent extensive inspection and testing to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.

4. Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation and that the vessel will not fail under the conditions of any of the postulated accidents.

5. Will be subjected to monitoring and periodic inspection in order to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

Response to Contention 3.3.12

The Staff, in its evaluation, recognized that reactor coolant pump overspeed in the event of loss-of-coolant accident can only be attained if the physical conditions allow acceleration of the pump components during the course of the transient. The Staff has identified many factors which could not only interfere with the overspeed transient but also prevent attainment of the stated flywheel overspeed level. Insufficient data and analyses are available to justify the premise that such pump overspeeds are attainable. Fragmentation of internal pump components, which may contain service-induced flaws would cause interruption of pump acceleration much before any significant missiles could be generated to cause damage to surrounding critical components.