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DESCRIPTION:

Ltr re our 10-10-73ltr....furnishing info concerning analysis of ATWS.....

ENCLOSURES:

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ACKNOWLEDGED

PLANT NAME: Kewaunee

FOR ACTION/INFORMATION 11-6-74 ehf

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November 1, 1974

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RECEIVED NOV 5 1974 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. Edson Case, Acting Director
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Case:

Subject: Docket 50-305
Operating License DPR-43
Anticipated Transients Without Scram

In accordance with the requirements of AEC letter to Mr. E. W. James, dated October 10, 1973 and in compliance with the recommendations set forth in WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," we submit the following:

The AEC Staff position presented in WASH-1270 required an analysis of certain operational transients, due to random or common mode failure, without operability of the redundant Reactor Protection System. The Kewaunee Reactor Protection System is designed incorporating equipment separation, both mechanical and electrical by channel, to prevent to the extent possible, common mode failure. Detailed procedures which have been verified to be functionally correct, during the preoperational test and startup test program, are implemented to prevent common mode failure by calibration. Section 7 of the Final Safety Analysis Report addresses the system design and WCAP-7486, "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," discusses the reliability of the system.

Although the occurrence of an incident corresponding to an anticipated transient without scram is extremely remote, the evaluation of the occurrence has been performed in compliance with the recommendations set forth in WASH-1270. The results of this evaluation are reported in Westinghouse Topical Report WCAP-8330, "Westinghouse ATWT Analysis." WCAP-8330 covers the analysis of ATWT as it applies to the Kewaunee Nuclear Power Plant. The 2-loop model described and the parameters used in the analyses conservatively bound those used for the Kewaunee Plant and are tabulated in the attached table.

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Sensitivity studies of the difference in the parameters, such as steam generator mass, have been performed and are described in WCAP-8330 and they demonstrate minimal effect on DNB margin or on Reactor Coolant System peak pressure. Since the 4-loop model analyses have been identified as the worst case for transients such as rod withdrawal from subcritical, uncontrolled RCCA bank withdrawal at power and station blackout, would result in more conservative results for the 2-loop analyses:

Differences between our unit and the 2-loop models used in WCAP-8330 and the effects on the analyses are:

(1) Increase in total Reactor Coolant System Volume - No significant effect on the analyses.

(2) Decrease in nominal steam flow - No significant effect on the analyses.

(3) Decrease in the auxiliary feed flow capacity - No significant effect, but a slightly higher peak Reactor Coolant System pressure.

In our review of WCAP-8330, we concur with the conclusions arrived at in the report which are:

(1) Because of the admittedly low probability of occurrence, DNB would be acceptable for an ATWT event. However, based on the calculated minimum DNB ratios, no significant clad damage is expected. This result stems from the nuclear-thermal-hydraulic design margin in current Westinghouse reactors.

(2) In no case does the Reactor Coolant System pressure exceed the maximum allowable pressure let alone approach the ultimate failure limits of the Reactor Coolant System. This result stems from the conservative approach to overpressure protection; safety valves and power-operated relief valves are sized without consideration for direct reactor trip or inherent shutdown characteristics of a PWR, although credit for these considerations is allowed by the ASME Code, Section III.

(3) The containment pressure transients yield peak pressures below design. These containment pressure transients are negligible compared to the containment design basis transient, e.g., loss-of-coolant accident.

(4) The core thermal performance, the volume of reactor coolant and secondary fluid released, and the containment pressure transient are all considerably less severe for these ATWT events than for the design basis accident and the dose consequences for ATWT are well within the guideline values set forth in 10 CFR 100.

Furthermore, it should be noted that although the analyses presented in WCAP-8330 verifies present design adequacy, the assumption that no operator action is taken during transients of the ATWT nature is extremely unrealistic. The most rapid transients are on the order of tens of seconds duration following initiation of the first automatic trip and on the order of minutes to water relief out of the pressurizer. When numerous reactor trip annunciators indicate that a trip should have occurred and the system parameters as indicated on the control board meters verify that the trip points have been exceeded without indication of a rod drop it is incredible to assume that licensed qualified operators would not recognize that an ATWT was in progress and take appropriate action. The rod drive MG sets are supplied power by a non-vital bus which can be de-energized from the control room without affecting the ability of the plant to function safely after a trip. The tripping of this bus is in effect a second manual trip mechanism which realistically should be acknowledged.

Based upon the review of WCAP-8330, we believe the analyses adequately demonstrate that no hardware modifications are required in the Kewaunee Plant to mitigate the consequences of an ATWT, therefore, no amendments to the FSAR is necessary.

Very truly yours,

E. W. James
Senior Vice President
Power Generation & Engineering

EWJ:sna

cc - Mr. Dwane Boyd, US AEC - Resident Inspector
Mr. J. G. Keppler, US AEC - Region III

ATWS MODEL PARAMETERS

PARAMETER	WCAP 8330 Model	Kewaunee
RC Pump Type	Model 93A	93A
Steam Generator Type	51 Series	51
No. of Loops	2	2
Core Power (MWt)	1644	1644
Nominal Pressurizer Pressure (PSIA)	2250	2250
Nominal Coolant Flow (gpm) total	178,000	178,000
Nominal Average Coolant Temperature (^{o}F)	567.3	567.3
Nominal Coolant No-Load Temperature (^{o}F)	547	547
Total RCS Volume Including Pressurizer (ft^3)	6230	6280.2
Pressurizer Volume (ft^3)	1021.3	1000.0
Steam Capacity of Power Operated Relief Valves lb/hr @ 2350 PSIA	2 at 210,000	2 at 210,000
Steam Capacity of Safety Valve lb/hr	3 at 325,000 @ 2500 PSIA	3 at 336,700 @ 2590 PSIA
Best Estimate Rod Worth of Bank D at its Full Power Insertion Limit (% $\Delta k/k$)	0.3	0.3
Steam Generator Design Pressure (PSIA)	1100	1100
Steam Generator Nominal Steam Temperature (^{o}F)	510.8	510.8
Nominal Steam Flow (lb/sec)	1996	1969.4
Nominal Fluid Mass in Steam Generator (lb)	203,200	217,000
Auxiliary Feedwater Temperature (^{o}F)	>130	<100
Auxiliary Feedwater Available (gal)	150,000	150,000 + L. Michigan
Capacity of Auxiliary Feedwater (gpm)	800	240 ea.*
Volume of Line Between Auxiliary Feedwater Connection on Feedline and Steam Generator Inlet, Total for all Loops (ft^3)	261	5.7
Nominal Feedwater Temperature (^{o}F)	435.8	427.3

* 3 pumps available