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FROM: Westinghouse Elect Corp.
Pittsburgh, Pa. 15230
M. H. Judkis

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Consists of responses to Outstanding items of NRC report to ACRS dtd 11/04/77 in the matter of the Donald C. Cook Nuclear Plant Unit #2...w/att supporting information......

3p + 5p

PLANT NAME: DONALD C COOK UNIT # 2
jcm 11/22/77

ENCLOSURE

40 ENCL *

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Power Systems Company
PWR Systems Division
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November 18, 1977

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U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Dear Mr. Case:

AMERICAN ELECTRIC POWER
Donald C. Cook Nuclear Plant Unit 2
Responses to Outstanding Items of NRC Report to ACRS

The NRC report to the Advisory Committee on Reactor Safeguards (dated November 4, 1977) in the matter of the Donald C. Cook Nuclear Plant Unit 2 has been reviewed with respect to the Outstanding Items listed in Section 1.8. This letter addresses those outstanding items in the Westinghouse area of cognizance. A separate letter from the American Electric Power Service Corporation will forward the authorization for this submittal on Docket Number 50-316. The responses to these outstanding items are given below in the same sequence that the concern appears in the report.

1. The additional information requested with regard to the new thermal design basis THINC-IV uncertainties has been provided in a Westinghouse letter, NS-CE-1583, October 25, 1977, from Mr. C. Eicheldinger to Mr. J. F. Stolz, NRC.

2. The additional information requested with regard to the WRB-1 heat transfer correlation has been provided in a Westinghouse letter, NS-CE-1581, October 24, 1977, from Mr. C. Eicheldinger to Mr. J. F. Stolz, NRC.

3. The additional information requested relative to the treatment of the radial pressure gradient, THINC-IV code has been provided in a Westinghouse letter, NS-CE-1591, November 2, 1977, from Mr. C. Eicheldinger to Mr. J. F. Stolz, NRC.

4. The steam generator subcompartment pressure response analysis in response to Question 22.3 will be submitted to you by January 3, 1978. The preliminary analysis was presented to the NRC in a meeting on November 11, 1977, and are provided in a separate letter. The preliminary results presented in the meeting show that the postulated accident does not endanger public health and safety.

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5. The additional documentation requested with regard to the electrical equipment environmental qualification tests is provided in a separate Westinghouse letter, AEW-7035, dated November 18, 1977, addressed to Mr. Edson G. Case.

6. The revision to the Westinghouse fuel rod performance code, PAD 3.3, described in WCAP-8720, "Improved Analytical Methods Used in Westinghouse Fuel Rod Design Calculations" has no impact on loss of coolant accident (LOCA) analysis results performed for Donald C. Cook Unit 2. The maximum stored energy occurs early in life when the PAD 3.3 and the approved PAD 3.1 codes give essentially equal fuel temperatures. In fact, equal peak average temperatures were assumed in the LOCA analyses. Calculations were also made for higher burnups where PAD 3.1 and 3.3 give different values of temperature and rod pressure. However, the early in life conditions were found to be most limiting.

   PAD version 3.1 gives higher values of fission gas release than PAD 3.3 during typical first and second cycles of operation. It is only for the third cycle and beyond that PAD 3.3 results in higher fission gas release rates than PAD 3.1. The lead rod will not be calculated to exceed system pressure during the first cycle of operation with either PAD version

   The modified design criterion for fuel rod internal pressure is:

   "The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady state operation; and (2) extensive DNB propagation to occur."

   In addition, we have reviewed the fuel damage assumptions used in the site dose evaluation for this plant, and we find that with the revised design criterion the dose consequences of the accidents remain essentially unchanged and well below the consequences due to the LOCA accident as reported in the FSAR.

7. The conclusion for an analysis of a postulated loss-of-coolant accident with the accumulators isolated with system pressure at 1000 psia and system temperature at 425°F has been presented in FSAR Amendment 78 in response to Question 212.33. Additionally, the peak clad temperature would be expected to occur at the core midplane, where maximum power is available. The clad temperature at the midplane will turn around shortly after BOC (bottom of core recovery) as it does in the SAR ECCS analysis, an additional increase of (4.7 x 6) 28°F is anticipated based on adiabatic heatup at the core midplane, giving a peak clad temperature of 1746°F.

8. A calculation for postulated gas blanketing on the tube side of the steam generators for breaks less than two inches has been made in addition to the response to Question 212.34 (FSAR Amendment 78). The maximum volume of hydrogen that could be available to accumulate in the steam generators is 12Ft^3 at 1200 psia and 567°F, which are the conditions at which the RCS would stabilize when the steam generator safety valves are actuated. This volume of non-condensible gas distributed among the four steam generators results in approximately a 0.27 foot (or ~0.1 psi) reduction in the available elevation head in each steam generator. For a long time period the core and hot leg side of the steam

generators are voided so that a significant driving head exists. When the RCS becomes water solid the density difference between the hot side of the RCS and the cold side of the RCS is sufficient to drive flow around the loop. The effect of non-condensibles accumulating in the steam generator would be to simply delay the time at which the system becomes water solid. Adequate core cooling will at all times be provided.

9. The behavior of the primary system pressure after a postulated steamline break has been evaluated. This evaluation is presented in the attached Appendix B to the response to Question 212.34.

10. Westinghouse is performing a feedwater line break analysis, which will be presented for NRC review in November 1977.

11. Analysis of containment temperature and pressure long term response to a postulated steamline break will be presented approximately five months following the NRC approval of the Westinghouse LOTIC-3 code. This commitment was made in the response to Question 22.9 presented in Appendix Q of the FSAR.

If there are any questions on the above, please contact this office.

Very truly yours,

M. H. Judkis, Manager
American Electric Power Project

I. C. Ratsep/lk
Attachment Forty (40)

cc: R. W. Jurgensen, 1L, 5A
R. S. Hunter, 1L
R. F. Hering, 1L
S. H. Horowitz, 1L
P. W. Daley, 1L
S. J. Milioti, 1L
J. G. Feinstein, 1L

APPENDIX B

The following additional information is provided as requested in regard to Question 212.34 (Amendment 78).

Regarding the concern that pressurizer thick metal could cause flashing of water entering the pressurizer during its refill phase following a hypothetical steamline break the following conservative calculation was made.

Assumptions:

1. The energy contained initially in the pressurizer wall area in contact with the refilling water was transferred to the water until the pressure would no longer permit flashing.

2. Only the portion of the pressurizer wall in contact with the fluid was assumed to transfer its stored heat.

3. Pressurizer heat loss through its insulation during the transient was not considered.

4. The water temperature used to calculate flashing was the highest entering the transient during the refill phase.

5. All heat was assumed to be transferred by the time of peak repressurization (600 seconds) following the break.

Physical Parameters: Pressurizer shell inside diameter = 84 in.

Pressurizer shell thickness = 4 in.

Pressurizer lower head inside radius = 43 in.

Pressurizer lower head thickness = 2.8 in.

Pressurizer metal density = 489 lbm/ft.^{3}
Pressurizer metal heat coapacity = 0.13 Btu/lbm°F

Pressurizer fluid enthalpy = 250 Btu/lbm
(Peak during refill)

Initial wall temperature = 652.7°F
(Tsat at 2250 PSIA)

The peak pressurizer water volume during the transient was 677.8 Ft^3. With a pressurizer lower head volume of 88 Ft^3 the water level would thus rise to 15.33 ft. of the cylindrical shell height. The amount of metal mass in contact with the fluid would be

$$\pi \left(R^{\text{shell}}_{\text{inside}}{}^2 - R^{\text{shell}}_{\text{outside}}{}^2\right) h_{\text{shell}} + \frac{2}{3}\pi \left(R^{\text{head}^3}_{\text{outside}} - {}^{\text{head}^3}_{\text{inside}}\right) = 138.5 \text{ Ft}^3$$

$\dot{M}Cp = 138.5\ Ft^3 * 489\ lbm/Ft^3 * 0.13\ Btu/lbm - °F = 8804\ Btu/°F$

The energy gained by the water boiled to make steam must be given up by the pressurizer metal wall. When the pressurizer wall temperature decreases to the saturation temperature at the system pressure boiling will cease.

ΔE pressurizer wall = $-\Delta E$ Boiled Steam

$$\dot{M}Cp\ \Delta T_{wall} = -\Delta M_{steam}\ (h_{fluid} - h_{steam})$$

By assuming a system pressure a ΔT_{wall} can be calculated and a solution for ΔM_{steam} can be found. By assuming an isentropic compression of steam in the pressurizer the new system pressure can be found by calculating the fraction of the steam space occupied by the original steam.

$$V_{Final} = \frac{M_{initial\ steam}}{M_{initial\ Steam} + \Delta M_{steam}} * V_{final}\ \text{before mass addition}$$

Once V_{final} is calculated P_{final} can be calculated by knowing the pressure and steam volume at the time of the beginning of pressurizer refill. At the beginning of refill the steam volume is just the pressurizer and surge line volume. At the beginning of refill the pressure is 605.8PSIA and the steam volume 1843.7 ft^3.

$$Po\ Vo^{1.4} = P_{final}\ V_{final}^{1.4}$$

$$P_{final} = Po\ \left(\frac{Vo}{V_{final}}\right)^{1.4}$$

The new P_{final} must be compared to the saturation pressure assumed in computing ΔT for the pressurizer metal. The solution is iterative and convergance is reached when P_{final} and the saturation pressure assumed in the ΔT calculation are identical. The final iteration of the calculation is given below.

Assume P_{final} = 1520PSIA T_{sat} = 598.0°F

$$Q = \dot{M}Cp\ (T_{initial} - T_{sat})$$

$$= 8804\ Btu/°F*(652.7°F - 598.0°F)$$

$$= 481579\ Btu$$

$$\Delta h = h_{sat\ steam} - h_{water}$$

$$= 1169.0\ Btu/lbm - 250\ Btu/lbm$$

$$= 919.\ Btu/lbm$$

$$\Delta M_{steam} = \frac{Q}{\Delta h} = \frac{481579 \text{ Btu}}{919 \text{ Btu/lbm}} = 524.0 \text{ lbm}$$

At Po. = 605.8PSIA, Vo = $1843.7Ft^3$

$$M_{initial\ steam} = 2416\text{lbm}$$

$$V_{final} = \frac{2416\text{lbm}}{2416\text{lbm} + 524\text{lbm}} * (1843.7\text{lbm} - 677.8\text{lbm})$$

$$P_{final} = 605.8 \left(\frac{1843.7}{958.1}\right)^{1.4} \text{PSIA}$$

$$= 1520\text{PSIA}$$

The calculated P_{final} equals the assumed P_{final} so the final pressure has converged. The final pressure calculated is approximately 500PSI higher than the peak pressure of 1020PSIA given in Figure 212.34-1 of the response to question 212.34 for the limiting case with respect to reactor vessel integrity. An evaluation of the reactor vessel integrity indicates that vessel integrity will be assured for the 40 year design life of the plant assuming the increased pressure.

The calculation presented is very conservative. No credit is taken for pressurizer metal heat loss to the containment through the pressurizer insulation. Neither is credit taken for the recondensation of steam bubbles fromed at the wall as they rise through the subcooled water in the pressurized. The calculation also assumes that boiling heat transfer is the mechanism for all heat removal from the wall. During later portions of the heat transfer the wall superheat will not support boiling. Therefore, the actual impact on peak pressure of the pressurizer wall metal is expected to be much less than the calculation presented.

Figure 212.34-21 presents the hot leg temperature in the intact loops of the reactor coolant system for a hypothetical large steam line break transient. At 200 seconds (the beginning of pressurizer refill) the hot leg temperature is approximately 280°F. This serves as the basis for the pressurizer fluid enthalpy of 250 Btu/lbm used in a conservative calculation of the pressurizer thick metal heat effect on coolant system repressurization following 9 steam line break.

Figure 212.34-21 corresponds to the case with full forced convection reactor coolant flow, presented in figures 212.34-1 - 212.34-4 (Amendment 78). A similar figure is not provided for the case with free convection reactor coolant flow. The evaluation of reactor vessel integrity performed for the response to question 212.34 indicated that a similar size initial flow depth of less than approximately 1/3 wall thickness would assure reactor vessel integrity for the design life of the plant. The additional pressure due to fluid flashing in the pressurizer is much greater for the case with forced convection in the reactor coolant system than for the case with free convection. The reason for the larger pressure increase is that the pressurizer refill is much greater for the former case (677.8 ft^3) than the latter (278.4 ft^3). The larger pressurizer refill allows more thick metal to come into contact with the fluid, thus causing more flashing.

An evaluation of the reactor vessel integrity was performed using the conservatively calculated reactor coolant system pressure increase assuming pressurizer fluid flashing. The fracture analysis utilized linear elastic fracture mechanics methods in the evaluation. The fracture mechanics analysis results in the determination of the minimum depth flows that might propagate during the large steam line break. The results of the fracture mechanics analysis are that a flow having a depth less than 1/4 of the reactor vessel wall thickness will not propagate during the large steam line break at end of plant life fluence levels. This flow depth is within the range of flow depths that would not be missed during manufacturing and in-service inspections of the reactor vessel. Therefore, no flow propagation will occur during the large steam line break.

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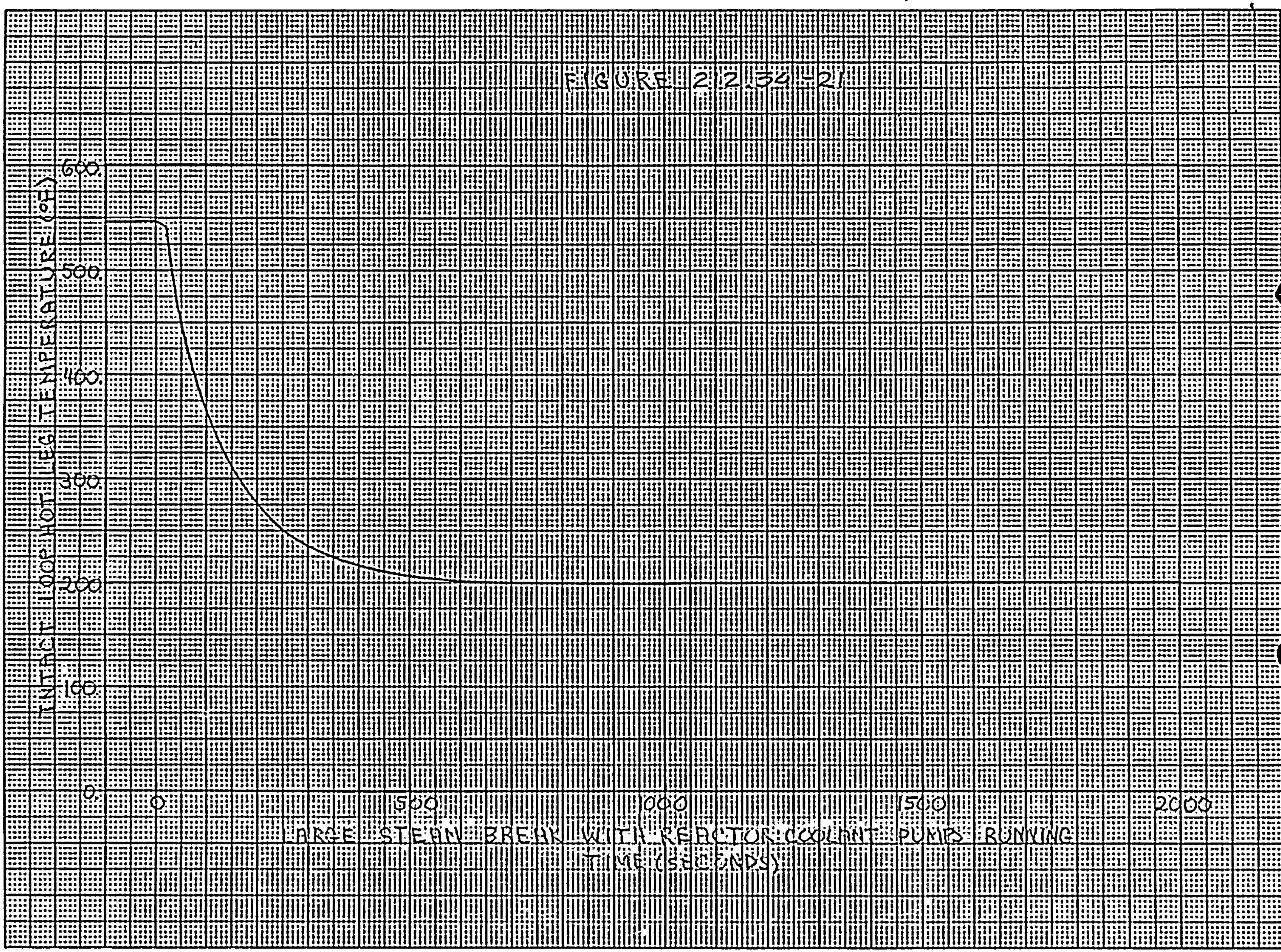

FIGURE 2.2.34-21

LARGE STEAM BREAK WITH REACTOR COOLANT PUMPS RUNNING