ATTACHMENT 1

Consumers Power Company
Big Rock Point Plant
Docket 50-155

Proposed Technical Specification
Page Changes

September 3, 1985

1 Page

5.2.2 Control Rod Performance

(a) Control Rod Performance

The following limits shall apply to any control rod which can be withdrawn. It shall be permissible to tag and valve out the hydraulic drive water to a fully inserted control rod which is defective or does not meet these limits provided the remaining rods do meet the limits.

The following tests shall be performed at each major refueling but not less often than once every 20 months, and prior to startup following any outage greater than 120 days in length.

(i) Withdrawal of each drive, stopping at each locking position to check latching and unlatching operations and the functioning of the position indication system.

(ii) Scram of each drive from full withdrawn position. Maximum scram time from system trip to 90 percent of insertion shall not exceed 2.5 seconds.

(iii) Insertion of each drive over its entire stroke with reduced hydraulic system pressure to determine that drive friction is normal.

(iv) Continuous withdrawal and insertion of each drive over its stroke with normal hydraulic system pressure.

(b) Core Shutdown Margin Verification

The reactivity of the core loading shall be such that it is always possible to maintain k_{eff} at less than 0.997 with the most valuable reactivity worth control blade completely withdrawn from the core. The core shutdown margin shall be verified by a demonstration that the reactor is subcritical with the most valuable reactivity worth control blade fully withdrawn, plus an immediately adjacent blade withdrawn to a position known to contribute 0.003 k_{eff} or more to the effective multiplication. In the event that the maximum reactivity condition occurs at a temperature greater than ambient, the demonstration will either be performed at that temperature or a suitable additional margin will be demonstrated at ambient.

This verification shall be performed prior to start-up after any shutdown in which the system has cooled sufficiently to

ATTACHMENT II

Consumers Power Company
Big Rock Point Plant
Docket 50-155

EVALUATION OF ROD DROP ACCIDENT IN SUPPORT OF
DELETION OF MINIMUM 23 SECOND WITHDRAWAL CRITERIA

September 3, 1985

23 Pages

TSOB0885-0005C-NL02

Standard Review Plan - 15.4.9

Acceptance Criteria

I. Reactivity excursions should not result in radially averaged fuel rod enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

II. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code.

III. The number of fuel rods predicted to reach assumed fuel failure thresholds, and associated parameters such as the amount of fuel reaching melting conditions, will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power and fuel cladding dryout for rated power initial conditions.

TABLE OF CONTENTS

SECTION I: METHODOLOGY USED TO MEET ACCEPTANCE CRITERION I

The peak deposited enthalpy resulting from a rod-drop accident was calculated according to the format outlined in Reference 1, pp. 11-12. Although Reference 1 shows a generic calculation of peak deposited enthalpy, it demonstrates the format followed for rod-drop accident analyses.

The basic assumptions made in the Exxon Report XN-NF-78-51 (Reference 1) are that:

1. A central core region control rod becomes decoupled from its drive mechanism and remains fully inserted in the core while the original withdrawal sequence continues.

2. At the worst point in the withdrawal sequence, the stuck rod becomes freed and falls out of the core, producing a prompt critical reactor which results in a rapid power increase that is effectively reduced to operational level through the Doppler feedback mechanism. As is shown in Figure 3.2 of Reference 1, scram bank insertion does not significantly affect the transient power level. However, the insertion of the scram bank, along with the Doppler feedback mechanism, does serve to bring the reactor down to a subcritical stage after Doppler feedback has reduced the power level to about 10 x (Maximum designed thermal power), or 2400 MWt.

3. The reactor remains at the subcritical stage due to the full insertion of the scram bank.

The X-TRAN computer code (Reference 3) was used to generate the graph of peak deposited enthalpy versus control rod worth for 4 different axial x radial peaking factors (Figure 1.1 of Reference 1). The base values for the Doppler coefficient and delayed neutron fraction used to generate Figure 1.1 are $\sigma_D = -9.52 \times 10^{-6}$ $\Delta k/k$-°F, and $\overline{\beta} = 0.00525$, respectively. Figures 1.2 and 1.3 of Reference 1 supply information necessary to adjust the peak deposited enthalpy (from Figure 1.1, Reference 1) according to different values of the Doppler coefficient and the delayed neutron fraction, respectively.

Specific analysis of a rod-drop accident was performed for Cycle 20 at Big Rock Point (Reference 4), and indicate that the total peak fuel enthalpy during the power transient is well under the specific energy design limit of 280 cal/g for UO_2 fuel (Reference 6). A total peak fuel enthalpy of not more than 179.95 cal/g would occur in the hottest core region during the power transient resulting from a rod-drop accident that occurs according to conditions outlined in Exxon report XN-NF-78-51. Appendix A contains a copy of Reference 4, and shows the conservative measures taken for the Cycle 20 rod-drop accident analysis. These measures include selection of a Doppler coefficient, and the assumption of an initial fuel enthalpy of 18 cal/g, which produce a slightly higher total peak fuel enthalpy than should actually occur.

SECTION II: METHODOLOGY USED TO MEET ACCEPTANCE CRITERION II

Some conservative assumptions regarding heat transfer and reactor vessel thermodynamics were made, and are listed below.

1. The energy generated in the core during the rod-drop transient power is instantaneously transferred from the fuel to the coolant at various times t_n, n=1,....13. This is very conservative since energy transfer delay time due to thermal resistance of the fuel, gap and cladding is considered to be zero.

2. The vessel is assumed to be enclosed by an adiabatic boundary, ie, no heat escapes from the reactor vessel during the rod-drop power transient. The system is thus considered to contain an adiabatic-reversible heat transfer mechanism for transient power.

3. Assumption #2 implies conservation of mass within the system, ie, initial coolant mass at the beginning of the power transient is equal to final coolant mass. The pressure relief safety systems therefore do not apply.

4. The mixture enthalpy (hm) as calculated in Table B-II is equal to h_f, the fluid enthalpy. This is conservative since mixture quality $q = (h_m - h_f)/(h_g - h_f)$. See Appendix B for details.

The maximum reactor pressure during the power excursion resulting from a rod-drop accident was calculated for a 10 mk control rod according to the format listed below. Appendix B contains details of the calculations involved at each step.

(i) The transient power curve (Figure 3.2, Reference 1) was integrated in increments from t_o = 0.55392 sec to t_n, n=1...13. The initial time t_o is taken as the point where reactor power increases above 240 MWt. This resulted in reactor core energy generation E_n after various time periods of the transient power level.

(ii) The enthalpy of the mixture (h_m) was calculated as initial enthalpy plus Δh_m. The quantity Δh_m is the enthalpy increase of the system due to the energy deposited from the rod-drop accident.

(iii) Mixture enthalpy of the system is taken as fluid enthalpy (h_f), and Reference 7 is used to calculate the corresponding saturation pressure and temperature, P_n and T_n, respectively.

CONCLUSION

As is shown in Table B-II, the vessel pressure increased to $\sim$ 1466 psia at t_{13} = 0.75 sec, producing a total pressure change of 116 psia due to the rod-drop power transient. Additional vessel pressure increase due to the power transient beyond 0.75 sec is insignificant, and was not calculated since the main steam bypass valve is fully open 0.2 sec after a pressure increase of 50 psia is attained (see Reference 8).

Therefore, the maximum vessel pressure during any portion of the power transient resulting from a rod-drop accident is considerably less than the design pressure of 1715 psia for the BRP reactor vessel. This is conservative relative to maximum design pressure according to "Service Limit C" of the ASME Code (Reference 2).

SECTION III - JUSTIFICATION FOR MEETING ACCEPTANCE CRITERION III

A generalized cladding failure threshold of 170 cal/g was established from a series of tests performed by J.E. Grund, et. al., in 1969/1970 (Reference 5), and given in General Electric report #NEDO-10527 (Reference 6). However, specific tests and analyses were performed for different fuel types in Reference 5, where GEP pellet fuel most closely matches Cycle 20 fuel at Big Rock Point. From Table III of Reference 5, the clad thickness for GEP fuel is 0.032 in., which compares quite well to the clad thickness of 0.034 in. for Big Rock Point H-type fuel.

Test results for GEP pellet fuel are shown in Table C-I, Reference 5, where CDC transient test No. 487 indicates cladding failure occurred at 205 cal/g. This test result should correspond more closely to the cladding failure threshold for H-type fuel. The generalized failure threshold of 170 cal/g (Reference 6) is derived from tests on different fuels, where clad thickness varied from 0.014 in. to 0.032 inches.

Reference 15 displays results of transient tests conducted on 0.3125 in. o.d. pellet fuel with a clad thickness of 0.020 in. and clad material consisting of Zr-2. Loss of clad integrity for this down-scaled fuel rod occurred at fuel enthalpies ranging from 223 to 256 cal/g. Although the higher clad failure threshold is expected due to the smaller fuel diameter, this demonstrates the dependence of clad failure during transients on parameters such as fuel diameter and cladding thickness.

The total peak fuel enthalpy from a rod-drop accident at Big Rock Point was calculated in Reference 4 as approximately 180 cal/g (See Appendix A), using some very conservative parameters. This peak fuel enthalpy should occur in a region consisting of the four assemblies clustered around the dropped control rod. If the generalized clad failure threshold of 170 cal/g is assumed, a limited amount of fuel rod perforation may occur in the region of the four clustered assemblies, if peak fuel enthalpy approaches 180 cal/g in certain rods. The radiological consequences of such an accident can be gauged by comparison to the "maximum credible accident" (MCA) considered in Section 13, Volume 1 of the Final Hazards Summary Report (Reference 13). Specifically, the case of a 10% core meltdown resulting from core spray reduction during the MCA is employed. The immediate concern of an accident which causes fuel rod perforation is release of gaseous fission products contained in the fuel rod plenum of irradiated fuel. For rod locations under going clad melting, the fuel pellets would fall out of the core, preventing these pellets from reaching melting temperatures. An overall summary of the radiological consequences of a 10% core meltdown is presented in Figures 13.5-13.15, Reference 13, and indicate that radiation doses are insignificant at the distance of a small population zone, and are not considered hazardous at distances near the plant boundaries. The only precaution that may be necessary is non-use of milk products for approximately 3-4 weeks due to ground deposition of Iodine-131 in an area of less than 1 sq. mi. near the plant site, as described on p.27, sec. 13, Reference 13. This implies that

the radiological consequences of a rod-drop accident would have an insignificant impact on the area near the plant boundaries, since highly localized clad melt temperatures would be generated, and would not approach 10% core meltdown. This assumes that fuel melting temperatures are not attained in the rod-drop accident.

Therefore, if a 170 cal/g clad failure threshold is assumed for current BRP fuel, the radiological impact of a rod-drop accident described by Exxon Report XN-NF-78-51 (Reference 1) and calculated in Reference 4 would still be considered negligible.

APPENDIX A

Determination of Peak Fuel Enthalpy for the BRP Reactor

The calculation of peak deposited enthalpy using the format of Exxon Report XN-NF-78-51 relies on determination of 5 neutronic parameters. These are:

1) Calculation of the hot standby dropped rod worth for a particular rod which will cause the reactor to reach a prompt critical stage. The reactor is assumed to be near a critical state initially (p.1, Reference 1) with a corresponding reactivity less than the delayed neutron fraction (p.3, Reference 1).

2) Calculation of the axial x radial peaking factor, excluding xenon, Doppler, and thermal-hydraulic feedback mechanisms.

3) Calculation of the Doppler coefficient.

4) Calculation of the delayed-neutron fraction in hot standby mode.

5) Selection of an appropriate assembly local peaking factor.

Calculation of parameters (1), (2) and (4) was performed using the GROK computer code (Reference 11), and apply directly to Cycle 20 operation at Big Rock Point. Since peak deposited enthalpy for a simulated Cycle 20 rod-drop accident was calculated in Reference 4, the details are not shown here. Instead, a copy of Design Review B*C*20*840828 has been included.

APPENDIX B

Determination of Incremental Vessel Pressure

Calculations listed below are ordered to correspond to the format steps of Section II.

(i) Equation of curve used to approximate reactor core power of Figure 3.2, Reference 1:

$\log P = mt + b,\ t_o \leq t \leq t_5$ and $t_7 \leq t \leq t_{12}$.

$\Rightarrow P = C10^{mt}$, where $C=10^b$, and m=slope of log P.

Equation of energy used to calculate reactor core energy of Figure 3.2, Reference 1:

$$E = \int P\,dt = \int_{t_i}^{t_n} C10^{mt}dt;$$ for regions 1 and 2 as defined below

$$\Rightarrow E_n = \frac{c}{m\ln(10)}\ [10^{mt_n} - 10^{mt_i}];$$ for regions 1 and 2 as defined below.

Results of graphical analysis of Figure 3.2, Reference 1 are shown in Table B-I, utilizing the equations below:

Region 1	Region 2
$0.55392 \leq t \leq 0.59373$	$0.60863 \leq t \leq 0.65$
$M_1 = 62.61384$	$M_2 = -40.85246$
$b_1 = -34.68306$	$b_2 = 27.39672$
$C_1 = 10^{-34.68306}$	$C_2 = 10^{27.39672}$

$$\therefore \quad E_n^1 = \frac{C_1}{m_1 \ln(10)} \; [10^{m_1 t n} - 10^{m_1 t 0}], \qquad E_n^2 = \frac{C_2}{m_2 \ln(10)} \; [10^{m_2 t n} - 10^{m_2 t 7}],$$

$n=1,2,3,4,5.$ $\qquad$ $n=8,9,10,11,12.$

For the time interval t_5 to t_7, a constant power level was assumed. This power level was calculated by using the power equations in Regions 1 and 2 at times t_5 and t_7, respectively, to find an average.

Thus, $P_{t_5-t_7} = \frac{P_1 + P_2}{2} \cong 326.0$, normalized,

where $P_1 = 10^{-34.68306} \; 10^{62.61384(0.59373)}$, Region 1

$P_2 = 10^{27.39672} \; 10^{-40.85246(0.60863)}$, Region 2

The truncated-triangle approximation to the logarithmic power increase of Figure 3.2, Reference 1 is thus conservative due to the constant peak power assumption for $t_5 < t < t_7$. The constant power assumption for $t_{12} < t \leq t_{13}$ is likewise conservative.

TABLE B-I

	Time, t_n (sec)	Δtn (sec)	*Normalized Power, Pn	**Normalized Energy, En(10^{-2})	**Energy Generated, En (MJ)
t_0	.55392	0	1.0	0	0
t_1	.56	.00608	2.403	.973	2.335
t_2	.57	.01608	10.158	6.352	15.245
t_3	.58	.02608	42.950	29.097	69.833
t_4	.59	.03608	181.60	125.263	300.631
t_5	.59373	.03981	---	214.966	515.918
t_6	.60	.04608	326.0	419.368	1006.483
t_7	.60863	.05471	---	700.706	1681.694
t_8	.61	.05608	299.72	744.531	1786.874
t_9	.62	.06608	117.00	938.777	2253.065
t_{10}	.63	.07608	45.674	1014.605	2435.052
t_{11}	.64	.08608	17.830	1044.206	2506.094
t_{12}	.65	.09608	6.960	1055.761	2533.826
t_{13}	.75	.19608	5.000	1105.761	2653.826

* (P_0=1 => 240 MWt)

**($E_n = E_{tot}^{tran} \mid_{t=tn}$)

TABLE B-II

	Time, tn (sec)	Δtn (sec)	ΔEn (MJ)	Mixture Enthalpy, hm(BTU/lb)	Saturation Temperature, Tn (°F)	Saturation Pressure, Pn (psia)	ΔPn (psia)
t_0	.55392	0	0	592.3	582.32	1350.0	---
t_1	.56	.00608	2.335	592.3	582.32	1350.0	0
t_2	.57	.01608	12.910	592.4	582.39	1350.8	0.8
t_3	.58	.02608	54.588	592.7	582.62	1353.1	2.3
t_4	.59	.03608	230.798	594.0	583.57	1363.1	10.0
t_5	.59373	.03981	215.287	595.2	584.45	1372.3	9.2
t_6	.60	.04608	490.565	598.0	586.49	1393.8	21.5
t_7	.60863	.05471	675.211	601.8	589.21	1423.1	29.3
t_8	.61	.05608	105.180	602.4	589.64	1427.7	4.6
t_9	.62	.06608	466.191	605.1	591.55	1448.5	20.8
t_{10}	.63	.07608	181.987	606.1	592.26	1456.2	7.7
t_{11}	.64	.08608	71.042	606.5	592.54	1459.2	3.0
t_{12}	.65	.09608	27.732	606.7	592.68	1460.8	1.6
t_{13}	.75	.19608	120.000	607.3	593.11	1465.6	4.8

Description of Variables:

(i) $\Delta t_n = t_n - t_o$, n=1...13.

(ii) $\Delta E_n = E_n - E_{n-1}$, n=1...13.

(iii) Mixture Enthalpy, $h_m = (h_m/_{t=to} + (\frac{En}{M_{tot}})$, where initial

mixture enthalpy, $h_m/_{t=to} = h_f = 592.3$ BTU/lb. The coolant mass total, M_{tot}=167,395 lbs., calculated for the BRP reactor vessel at 0% void, 582.32°F and 1350 psia (See References 7, 12). For $t_o < t \leq t_{13}$, h_m is conservatively taken as h_f, since $h_m > h_f$.

(iv) Both T_n and P_n were calculated using reference 7, and correspond to mixture enthalpy, since $h_m \approx h_r$.

(v) $\Delta P_n = P_n - P_{n-1}$.

REFERENCES

1) "Control Rod Drop Accident Analysis for Big Rock Point", Report No. XN-NF-78-51, Exxon Nuclear Company, Inc., January, 1979.

2) "ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components", 1983, Division I, subsection NC.

3) "XTRAN-PWR: A Computer Code for the Calculation of Rapid Transients in Pressurized Water Reactors with Moderator and Fuel Temperature Feedback", Report No. XN-CC-32, by J.N. Morgan, Exxon Nuclear Company, Inc., September, 1975.

4) Design Review B*C*20*840828, UFI-740/22*13*32; by S.M.Soltis, August 28, 1984.

5) "Subassembly Test Program Outline for FY 1969 and 1970", J.E. Grund, R.L. Johnson, K.O. Johnson, R.W. Miller, R.T. Johnson, B.E. Norton; IN-1313, Idaho Nuclear Corp., Idaho Falls; August, 1969 (IDO-17277).

6) "Rod Drop Accident Analysis for Large Boiling Water Reactors", C.J. Paone, R.C. Stirn, J.A.Woolley; Report No. NEDO-10527, 72 NED 18; Atomic Power Equipment Department, General Electric Company; March, 1972.

7) "ASME Steam Tables - Fourth Edition," C.A. Meyer, R.B. McClintock, G.J. Silvestri, R.C Spencer, Jr., Copyright 1979.

8) "Sequence of Pressure Relief During Transient", Letter from S.A. Bartosik to S.M. Soltis regarding procedure and results of "Turbine Bypass Valve System Functional Test", implemented 7/9/84 at BRP; August 21, 1984.

9) "Nuclear Reactor Analysis" J.J. Duderstadt, L.J. Hamilton, The University of Michigan, Copyright 1976 by John Wiley & Sons, Inc.

10) "Nuclear Engineering Handbook", Harold Etherington, Editor; Copyright 1958 by McGraw-Hill Book Company, Inc.

11) "Flare - A Three-Dimensional Boiling Water Reactor Simulator", D.L. Delp et.al., Report No. GEAP-4598; July 16, 1964.

12) "Big Rock Point Plant Technical Specifications" Docket 50-155, License DPR-6; Consumers Power Company; Amended May 18, 1984.

13) "Final Hazards Summary Report for Big Rock Point Plant", Volumes I and II, Consumers Power Company; November 14, 1961.

14) "Big Rock Point Fuel Data Book", According to Operational Reactor Physics Procedure ORP-B-05, updated 10/19/83; Consumers Power Company.

15) "The Response of UO_2 Fuel Rods to Power Bursts. Detailed Tests on 5/16-Inch OD, Pellet Fuel, Zircaloy Clad Rods." W.G. Lussie; IN-ITR-112, Idaho Nuclear Corp., Idaho Falls; January, 1970.

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DESIGN REVIEW SIGNOFF

Purpose: Perform a calculation of peak deposited enthalpy according to the total sequence of events postulated by Exxon report XN-NF-78-51. The calculation is performed for Big Rock Point, cycle 20, and supersedes the previous calculation.

Procedure Utilized: See Attachment #1.

Similarities With Previous Designs: None.

Summary of Results: The calculation of peak deposited enthalpy plus initial fuel enthalpy yielded a total fuel enthalpy of 179.95 cal/g. This value is well under the specific energy design limit of 280 cal/g for UO_2 fuel, despite the conservative values selected for initial fuel enthalpy, Doppler coefficient and assembly local peaking factor.

Special Media Attached (Drawings, Microfiche, etc.)

☐ No ☒ Yes - Include A List Of Attachments

Prepared By: S. M. Soltis | Date: 8/28/84

Reviewer's Comments: None
MLR

Reviewed By: Mike Reed | Date: 9/5/84

Approved By: T. E. Hallowell | Date: 9/7/84

B*C*20*840828

List of Attachments, Microfiche and References

Attachments

1.) Procedure for determination of peak deposited enthalpy resulting from a rod-drop accident.
2.) Calculation details of peak deposited enthalpy, and total deposited enthalpy.
3.) Checklist

Microfiche

1.) BOC 20 Rod Drop Accident - steps #34-36 DROP2ML

2.) BOC 20 Rod Worths #34-36 for Rod Drop WORTHGG
3.) BOC 20 Rod Drop Accident - Step #37 DROP2AP
4.) BOC 20 Rod Worths #37 for Rod Drop WORTHCH

References

1.) XN-NF-78-51, "Control Rod Drop Accident Analysis for Big Rock Point", Exxon Nuclear Company, Inc., January, 1979.
2.) Design Review B*C*20*840709, UFI-740/22*13*32; Big Rock Point Cycle 20 Final Physics Package.
3.) Design Review B.C.15.780520, by JA Umbarger, May 20, 1978.
4.) J.E. Grund, et. al., Subassembly Test Program Outline for FY 1969 and FY 1970, IN-1313, August, 1969 (IDO-17277).
5.) Design Review B*C*20*840430, UFI-740/22*13*32; Big Rock Point Plant Preliminary Physics Package for Cycle 20.
6.) Design Review B.Z.GROK.830304, by ML Reed, March 4, 1983

Attachment #1

Procedure for determination of peak deposited enthalpy resulting from a rod-drop accident

The procedure utilized herein follows the format established in reference 1, pp. 11-12. This calculation of peak deposited enthalpy was performed in order to insure that the upper limit of 280 cal/g total fuel enthalpy is not violated when the overall sequence of events postulated on p.3 of reference 1 are considered. The 280 cal/g fuel enthalpy limit is derived from reference 4.

Initially, the reactor is assumed near a critical state. Interpretation of postulates 4 and 5 on p.3 of ref. 1 will indicate that a prompt critical reactor state is attained after the stuck rod becomes free and falls out of the core. This criterion is met if a central core region control rod at step #37 of the BRP BOC 20 withdrawal sequence is considered, ie, a central core region control rod is assumed stuck initially and falls out of the core at step #36 of the withdrawal sequence.

The possibility of reaching a prompt critical state from steps 34, 35 and 36 of the withdrawal sequence based on a dropped rod at hot standby conditions was also considered. See attachment 2 for Details. Ideally, the steps analyzed should have control rods in the central core region at notch 23, as was the case for cycle 20 at Big Rock Point.

The axial x radial peaking factor was calculated by making a GROK computer-code run in the SCRAM mode (fiche #1,3), which obviates the xenon, Doppler and thermal-hydraulic feedback mechanisms as required by ref. 1. This run was performed for withdrawal

sequence step 37 as well as steps 34, 35 and 36.

Conservative measures were taken in the selection of a low-power Doppler coefficient, and assembly local peaking factor. The low-power Doppler coefficient of -9.267×10^{-6} $\Delta k/k/°F$ is taken from p. 1 of the BOL case for G3 fuel at 0% void and at a fuel temperature of 1377°C, in ref. 3. The assembly local peaking factor of 1.17 is taken from fiche #15 of ref. 6, which is a documentation of physics constants for H3 type fuel. Fiche #15 (of ref. 6) is a listing of local peaking factors for H3 at 0% void, and matches the assumed core conditions in ref 1.

Attachment #2 P. 1 of 2

Calculation details of peak deposited enthalpy, and total deposited enthalpy

By definition, a reactor is in the prompt critical state when:

$$\rho \geq \beta \; ; \quad \text{where } \rho = \frac{k - k_{crit}}{k}, \; k = k_{in},$$

and β = delayed neutron fraction.
k_{crit} = 0.989834 (ref. 5)

Using data from fiche #2 & Fiche #4:

	STEP 34	STEP 35	STEP 36	STEP 37
Peak Hot Standby Rod Worth, % $\Delta k/k$	0.64	0.58	0.46	0.44
* Hot Base Eigenvalue, k_{in} =	0.982700	0.986755	0.995117	0.997989
$\Rightarrow k_{out}$ =	0.976411	0.981032	0.990539	
Delayed Neutron Fraction, β_{hot}	0.005681	0.005714	0.005781	0.005792
Reactivity, ρ	-0.007260	-0.003120	0.005309	0.00817

Rod Drop Accident Parameters

The appropriate parameters for step 37 of the withdrawal sequence are listed below.

i) Peak Hot Standby Rod Worth ($10^{-3}\Delta k$)	4.39
ii) Axial x Radial Peaking Factor	3.99453
iii) Doppler Coefficient ($\Delta k/k/°F$)	-9.267×10^{-6}
iv) Total Delayed Neutron Fraction, β_{hot}	0.005792
v) Assembly Local Peaking Factor	1.17

*(Using the basic rod worth definition, $\left|\frac{k_{out} - k_{in}}{k_{in}}\right| = RW$)

Attachment #2

The rod worth conversion from $\% \Delta k/k$ to $10^{-3} \Delta k$ proceeds as follows:

$$\text{Rod Worth, } 10^{-3}\Delta k = \left[RW_{\text{dropped}}, \% \tfrac{\Delta k}{k}\right] \cdot \left[k_{in}\right] \cdot \left[10\right]$$

Thus, for step 37:

$$\left[0.44 \% \Delta k/k\right] \cdot \left[0.997989\right] \cdot \left[10\right] \cong 4.39 \quad 10^{-3}\Delta k = 4.39 \text{ mk}$$

Using fig. 1.1 of ref. 1, interpolation between the axial x radial PF curves of 5.00 and 3.70 yields a p.d.e. (peak deposited enthalpy) of approximately 135.30 cal/g. Linear interpolation in such a low rod worth region should be relatively accurate.

Using fig. 1.2 of ref. 1, p.d.e. multiplier = 1.045, Doppler.
Using fig. 1.3 of ref. 1, p.d.e. multiplier = 0.979, β_{hot}.

Adjusting the p.d.e. for the Doppler coefficient and delayed-neutron fraction:

$$\text{Adjusted p.d.e.} = (135.30)(1.045)(0.979) \text{ cal/g} \cong 138.42 \text{ cal/g}.$$

Further adjustment of the p.d.e. to account for local peaking effects in an assembly yields:

$$\text{L.P.-Adjusted P.D.E.} = (138.42)(1.17) \text{ cal/g} \cong 161.95 \text{ cal/g}.$$

Assuming an initial fuel enthalpy of 18 cal/g as ref. 1 suggests, total deposited enthalpy in the fuel, resulting from a rod-drop accident as described in attachment 1, is approximately 179.95 cal/g peak.

Attachment #3

Checklist

1.) Does the procedure described in attachment 1 agree with the rod-drop accident conditions postulated by reference 1? Yes, MLR

2.) Has the proper GROK execute tape been utilized in making the calculations on fiche-1 and 2?
Yes, MLR

3.) Are the calculations performed in attachment 2 correct? Yes, MLR

4.) Are the values assumed for the Doppler coefficient, assembly local peaking factor, and initial fuel enthalpy conservative?
Yes, MLR