Department of Energy
Washington, D.C. 20545

Docket No. 50-537
HQ:S:82:055

JUN 25 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated April 30, 1982

This letter formally responds to your request for additional information contained in the reference letter.

Enclosed are responses to Questions CS760.24, 29, 41, 45, 77, 79, 98, 134, 137, and 138; which will also be incorporated into the PSAR Amendment 69; scheduled for submittal later in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

cc: Service List
Standard Distribution
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D001

Question CS760.24

Please provide decay heat data as a function of core region and lifetime with their respective uncertainties.

Response

It should be noted that Section 15.1.4 (Effect on Design Changes on Analyses of Accident Events) has been added to the original PSAR to reflect the effect of design changes. Section 15.3 has not been changed in technical content since it was originally written in 1974. The worst case Section 15.3 undercooling transient has been updated and the analyses incorporated into Section 15.1.4.1. The current decay heat rates used in this update are included in the following information. If the 1974 decay heat data is of interest, it too can be supplied upon request.

The decay heat data used in the undercooling design event analysis presented in PSAR Section 15.1.4 are provided in Tables QCS760.24-1 through -5. Data are presented as a function of time after shutdown including uncertainty with the associated uncertainty value provided for each time point. Data are provided for the specific assemblies selected for hot channel analysis (see PSAR Figure 4.3-3). Decay times ranging from shutdown out to 500 seconds after shutdown were considered. The decay heat values are based on the heterogeneous core design in which the fuel and inner blanket assemblies have reached 2-year burnup and the radial blanket assemblies have a 3.2-year burnup.

Table QCS760.24-1

CLINCH RIVER SINGLE ASSEMBLY DECAY POWER VALUES W/UNCERTAINTIES

CRBRP HETEROGENEOUS CORE FUEL ASSEMBLY NO. 52

TIME AFTER SHUTDOWN (SECONDS)	DECAY POWER (KILOWATTS)	UNCERTAINTY (PERCENT)
0.	2.682E+02	3.242E+01
2.0000E+00	2.379E+02	2.742E+01
4.0000E+00	2.201E+02	2.507E+01
6.0000E+00	2.090E+02	2.370E+01
8.0000E+00	2.009E+02	2.274E+01
1.0000E+01	1.945E+02	2.200E+01
1.5000E+01	1.826E+02	2.065E+01
2.0000E+01	1.739E+02	1.967E+01
3.0000E+01	1.615E+02	1.829E+01
4.0000E+01	1.527E+02	1.729E+01
6.0000E+01	1.404E+02	1.592E+01
8.0000E+01	1.319E+02	1.496E+01
1.0000E+02	1.255E+02	1.426E+01
1.2000E+02	1.205E+02	1.373E+01
1.4000E+02	1.165E+02	1.334E+01
1.6000E+02	1.132E+02	1.305E+01
1.8000E+02	1.104E+02	1.283E+01
2.0000E+02	1.080E+02	1.266E+01
2.2000E+02	1.059E+02	1.254E+01
2.4000E+02	1.041E+02	1.244E+01
2.6000E+02	1.024E+02	1.237E+01
2.8000E+02	1.009E+02	1.232E+01
3.0000E+02	9.956E+01	1.229E+01
4.0000E+02	9.409E+01	1.218E+01
5.0000E+02	8.993E+01	1.210E+01

Table QCS 760.24-2

CLINCH RIVER SINGLE ASSEMBLY DECAY POWER VALUES W/UNCERTAINTIES

CRBRP HETEROGENEOUS CORE FUEL ASSEMBLY NO. 101

TIME AFTER SHUTDOWN (SECONDS)	DECAY POWER (KILOWATTS)	UNCERTAINTY (PERCENT)
0.	3.393E+02	3.231E+01
2.0000E+00	3.011E+02	2.732E+01
4.0000E+00	2.786E+02	2.497E+01
6.0000E+00	2.647E+02	2.361E+01
8.0000E+00	2.545E+02	2.264E+01
1.0000E+01	2.464E+02	2.191E+01
1.5000E+01	2.313E+02	2.056E+01
2.0000E+01	2.203E+02	1.958E+01
3.0000E+01	2.047E+02	1.820E+01
4.0000E+01	1.936E+02	1.721E+01
6.0000E+01	1.781E+02	1.583E+01
8.0000E+01	1.673E+02	1.488E+01
1.0000E+02	1.593E+02	1.417E+01
1.2000E+02	1.530E+02	1.365E+01
1.4000E+02	1.479E+02	1.326E+01
1.6000E+02	1.437E+02	1.297E+01
1.8000E+02	1.402E+02	1.275E+01
2.0000E+02	1.372E+02	1.258E+01
2.2000E+02	1.345E+02	1.246E+01
2.4000E+02	1.322E+02	1.237E+01
2.6000E+02	1.301E+02	1.229E+01
2.8000E+02	1.282E+02	1.224E+01
3.0000E+02	1.265E+02	1.221E+01
4.0000E+02	1.196E+02	1.209E+01
5.0000E+02	1.143E+02	1.201E+01

Table QCS760.24-3

CLINCH RIVER SINGLE ASSEMBLY DECAY POWER
VALUES W/UNCERTAINTIES

CRBRP HETEROGENEOUS CORE INNER BLANKET ASSEMBLY NO. 99

TIME AFTER SHUTDOWN (SECONDS)	DECAY POWER (KILOWATTS)	UNCERTAINTY (PERCENT)
0.	1.972E+02	2.871E+01
2.0000E+00	1.728E+02	2.349E+01
4.0000E+00	1.593E+02	2.100E+01
6.0000E+00	1.513E+02	1.954E+01
8.0000E+00	1.455E+02	1.851E+01
1.0000E+01	1.411E+02	1.772E+01
1.5000E+01	1.324E+02	1.626E+01
2.0000E+01	1.270E+02	1.522E+01
3.0000E+01	1.189E+02	1.376E+01
4.0000E+01	1.132E+02	1.274E+01
6.0000E+01	1.053E+02	1.133E+01
8.0000E+01	9.985E+01	1.036E+01
1.0000E+02	9.580E+01	9.650E+00
1.2000E+02	9.262E+01	9.118E+00
1.4000E+02	9.006E+01	8.724E+00
1.6000E+02	8.793E+01	8.421E+00
1.8000E+02	8.613E+01	8.188E+00
2.0000E+02	8.457E+01	8.005E+00
2.2000E+02	8.320E+01	7.861E+00
2.4000E+02	8.198E+01	7.747E+00
2.6000E+02	8.088E+01	7.660E+00
2.8000E+02	7.989E+01	7.588E+00
3.0000E+02	7.897E+01	7.536E+00
4.0000E+02	7.521E+01	7.341E+00
5.0000E+02	7.228E+01	7.211E+00

Table QCS760.24-4

CLINCH RIVER SINGLE ASSEMBLY DECAY POWER
VALUES W/UNCERTAINTIES

CRBRP HETEROGENEOUS CORE RADIAL BLANKET POSITION NO. 201

TIME AFTER SHUTDOWN (SECONDS)	DECAY POWER (KILOWATTS)	UNCERTAINTY (PERCENT)
0.	1.196E+02	2.936E+01
2.0000E+00	1.050E+02	2.421E+01
4.0000E+00	9.684E+01	2.177E+01
6.0000E+00	9.200E+01	2.035E+01
8.0000E+00	8.854E+01	1.934E+01
1.0000E+01	8.585E+01	1.857E+01
1.5000E+01	8.090E+01	1.715E+01
2.0000E+01	7.738E+01	1.615E+01
3.0000E+01	7.244E+01	1.473E+01
4.0000E+01	6.899E+01	1.375E+01
6.0000E+01	6.422E+01	1.239E+01
8.0000E+01	6.093E+01	1.147E+01
1.0000E+02	5.848E+01	1.079E+01
1.2000E+02	5.656E+01	1.029E+01
1.4000E+02	5.500E+01	9.912E+00
1.6000E+02	5.372E+01	9.626E+00
1.8000E+02	5.262E+01	9.409E+00
2.0000E+02	5.168E+01	9.245E+00
2.2000E+02	5.085E+01	9.117E+00
2.4000E+02	5.011E+01	9.014E+00
2.6000E+02	4.945E+01	8.933E+00
2.8000E+02	4.884E+01	8.871E+00
3.0000E+02	4.829E+01	8.825E+00
4.0000E+02	4.601E+01	8.661E+00
5.0000E+02	4.423E+01	8.549E+00

Table QCS760.24-5

CLINCH RIVER SINGLE ASSEMBLY DECAY POWER
VALUES W/UNCERTAINTIES

CRBRP HETEROGENEOUS CORE RADIAL BLANKET POSITION NO. 203

TIME AFTER SHUTDOWN (SECONDS)	DECAY POWER (KILOWATTS)	UNCERTAINTY (PERCENT)
0.	9.938E+01	2.854E+01
2.0000E+00	8.757E+01	2.351E+01
4.0000E+00	8.096E+01	2.111E+01
6.0000E+00	7.704E+01	1.972E+01
8.0000E+00	7.423E+01	1.873E+01
1.0000E+01	7.205E+01	1.797E+01
1.5000E+01	6.803E+01	1.660E+01
2.0000E+01	6.516E+01	1.561E+01
3.0000E+01	6.114E+01	1.423E+01
4.0000E+01	5.832E+01	1.326E+01
6.0000E+01	5.442E+01	1.194E+01
8.0000E+01	5.173E+01	1.104E+01
1.0000E+02	4.972E+01	1.038E+01
1.2000E+02	4.814E+01	9.897E+00
1.4000E+02	4.687E+01	9.531E+00
1.6000E+02	4.581E+01	9.258E+00
1.8000E+02	4.491E+01	9.047E+00
2.0000E+02	4.413E+01	8.883E+00
2.2000E+02	4.344E+01	8.758E+00
2.4000E+02	4.283E+01	8.654E+00
2.6000E+02	4.228E+01	8.575E+00
2.8000E+02	4.177E+01	8.518E+00
3.0000E+02	4.131E+01	8.465E+00
4.0000E+02	3.941E+01	8.309E+00
5.0000E+02	3.791E+01	8.193E+00

Question CS760.29

The intent of Chapter 15, Section 3 is to demonstrate the adequacy of the main heat transport systems to remove reactor and plant generated heat during protected undercooling accident conditions. Our concerns include:

Once a transient event is initiated there are several factors that could affect the outcome. In Section 15.3, these possibilities are touched upon, but not with any real consistency. Our assessment of important factors, and how they may be expected to vary, is organized in Table 15.3-2. It is implicitly assumed throughout the PSAR that one of the plant protection systems can fail to recognize a problem. Therefore, these transients must be analyzed with the more conservative signal, i.e., the one that leads to more severe conditions. The number of pony motors that come on line is certainly an important variable, especially since a pony motor driven pump in one loop could adversely affect the natural circulation in the other loops. Auxiliary feedwater could be supplied by two diesel driven pumps, one (50%) diesel driven pump (if one is out for service and turbine driven one fails), one turbine driven pump (which draws steam from the system), or none at all. The protected air-cooled condensers remove heat by natural draft air circulation. When they are needed, louvers are opened and fans come on, supposedly. How these various modes of operation impact on the undercooling events must be addressed.

Provide the basis and analyses to support the position that all events listed in Table 15.3-2 have been addressed on a consistent basis. Note that for several of the events, only a limited number of cases may need to be considered. In general, these are transients that are limiting only in the short term, i.e., in the longer term other related transients are likely to be more severe. Therefore, the events listed in Table 15.3-3 may need be analyzed only for the plant protection system and number of loops initially operating cases. However, at least six events should be analyzed for all reasonable cases. This is because all could lead to difficulties in long term heat removal, and bound some of the other events. These events are listed in Table 15.3-4.

TABLE 15.3-2 UNDERCOOLING EVENT CASES

Plant Protection System Available		(PPS)
Primary Shutdown System		
Secondary Shutdown System		
Pony Motor Pumps Available		
0, 1, 2 or 3 in Primary Loops		
0, 1, 2 or 3 in Intermediate Loops		
Auxiliary Feedwater		(AFW)
Both Diesel Driven	(100%)	
One Diesel Driven	(50%)	
One Turbine Driven	(100%)	
Protected Air Cooled Condensers		(PACCs)
Natural Draft, Louvers Closed	(0, 1, 2 or 3 loops)	
Natural Draft, Louvers Open	(0, 1, 2 or 3 loops)	
Fans On, Louvers Open	(0, 1, 2 or 3 loops)	
Number of Loops Initially Operating		
3-Loop Operation		
2-Loop Operation		

TABLE 15.3-3 LIMITED CASE ANALYSIS EVENTS

1. Spurious Primary Pump Trip
2. Spurious Primary and Intermediate Pump Trip
3. Spurious Intermediate Pump Trip
4. Inadvertent Closure of an Isolation Valve
   Evaporator Inlet
   Superheater Inlet
   Superheater Outlet
5. Turbine Trip
6. Inadvertent Actuation of Na/H_2O Reaction System
7. Single Primary Pump Seizure
8. Single Intermediate Pump Seizure
9. Small Water-to-Sodium Leaks in Steam Generator Tubes
10. Primary Heat Transport System Leak
11. Intermediate Heat Transport System Pipe Leak
12. Loss of One Recirculation Pump

TABLE 15.3-4 FULL CASE ANALYSIS EVENTS

1. Station Blackout (LOEP)
2. Loss of Normal Feedwater
3. Failure of the Steam Bypass System
4. Steam- or Feed- Line Break
   Main Steam Line Rupture
   Steam Line from Superheater to Steam Header
   Saturated Steam Line from Steam Drum to Superheater
   Feedwater Line Break
   Recirculation Line Break
5. Dump of Evaporatory Water Inventory with Inlet Isolation
   Valve Failure (Open)
6. Large Water-to-Sodium Leaks in Steam Generator Tubes

Response

The evaluation of the undercooling design events is discussed in Section 15.1.4.2 of the PSAR. As noted in that section, the set of events listed in Table 15.3-1 which were analyzed earlier were examined to determine the limiting undercooling event which turned out to be the loss of offsite power.

The question expresses a concern that all the factors (cases) listed in Tables 15.3-2 were not considered in the evaluation of events given in Tables 15.3-3 and 15.3-4.

Each of the events listed in Table 15.3-1 of the PSAR which involved a scram did consider the effects of primary system shutdown only, as well as only the secondary system shutdown. In addtion, it was further assumed that the highest worth rod in each of the shutdown systems was stuck in the out position. This provided a consistent basis for inclusion of a single failure in addition to the initiating event.

Some general observations about the thermal-hydraulic response of the CRBRP are in order:

1. The peak temperatures in the core (the real basis for the evaluation of udnercooling events) are in general seen immediately (within 20 seconds) after the onset of the event. The exact magnitude of these temperatures will be a function of the control rod worths, the delays in reactor scram and the reductions in primary flow prior to rod insertion. This is why the loss of offsite power is the limiting event for that list given in Table 15.3-1 of the PSAR.

2. For the unique case of loss of all primary pony motors, (the natural circulation event) the peak core temperatures will be seen between 200 and 300 seconds after scram after which the power to flow ratio begins to decrease and the core temperatures likewise decrease.

3. One primary pony motor will furnish more core flow than the case of no primary pony motors (natural circulation). Two primary pony motors will furnish more flow than the case of only one operating primary pony motor even if the check valve fails open in the loop with the inoperable pony motor. Thus, the case of no primary pony motor represents the limiting case.

4. Operation of pony motors in the intermediate loops (with no operation of primary pump pony motors) will enhance the primary natural convection flows because of the faster shift in the primary sodium temperatures in the IHX. Thus, this is not a limiting case.

5. Upsets in the steam generator system will not affect the peak core temperatures because of the long transport delays in the primary and intermediate piping. For example, the evaporator sodium outlet transient produced by a steam or feedline break in the shortest loop (loop 2) would require more than 200 seconds to be seen at the reactor vessel inlet even if the pony motor speeds in the primary and intermediate loops were 10% of rated flow. The effect of the transient produced in the affected loop would be further mitigated at

the core inlet due to the mixing of flows from the unaffected loops in the large mixing volume in the reactor inlet. The total decay power at 200 seconds would be less than 3.3%. Consequently, events 4, 6, 9 and 12 of Table 15.3-3 of the question are not limiting in the short term.

6. Events which affect the heat sinks for all three heat transport loops (and associated steam generator systems), given in Tables 15.3-3 and 15.3-4 of the question are:

   a) Turbine trip - see Section 15.3.1.5 of the PSAR.
   b) Station Blackout - Natural Circulation Analysis provided in CRBRP-ARD-0308.
   c) Failure of the Steam Bypass System - see Section 15.3.2.4 of the PSAR.
   d) Steam of Feedline Break - see Section 15.3.3.1 of the PSAR; Main Steam Line Rupture and Feedwater Line Break.

   As noted in the appropriate sections of the PSAR, none of these events result in signficant peak core temperatures.

7. Events which affect the heat sinks of individual loops (events 4, 6, 9, 12 of Table 15.3-3 and events 4 (except Main Steam Line rupture), 5 and 6 of Table 15.3-4) will not affect the peak core temperatures for the reasons given in 5 above. In the long term, the plant is fully capable of removing decay power through a single loop. Loss of a single loop due to the postulated events will not challenge the plants' decay heat removal capability; and in terms of peak core temperatures, would not represent a true undercooling event.

The factors (cases) provided in Table 15.3-2 of the question were considered and are discussed below:

> "Plant Protection System Available (PPS)
> Primary Shutdown System
> Secondary Shutdown System".

The limiting event in Section 15.3, Loss of offsite power, was analyzed for the secondary shutdown system only (PSAR pages 15.1-127 and 128). In addition, the hot rod analysis of the natural circulation event given in CRBRP-ARD-0308 also assumed a secondary shutdown system only. Results of analyses of the other events reported in Section 15.3 of the PSAR are given for both primary shutdown system only and secondary shutdown system only, where this aspect is important to the event being analyzed.

> "Pony Motor Pumps Available
> 0, 1, 2 or 3 in Primary Loops
> 0, 1, 2 or 3 in Intermediate Loops."

The limiting case is that in which it is assumed that there are no pony motors available, Event 1 of Table 15.3-4 of the question. This case has been reported in the natural circulation assessment (CRBRP-ARD-0308).

The combination of a primary pump seizure along with failure of the primary pump pony motors in the other two loops would be beyond the design base and has not, therefore, been considered.

It should be noted that the pony motors do not "come on line". They operate continuously. The load is picked up by the pony motors when the shaft speed reduces to the point where the over-running clutch engages. In addition, the two pumps in the same loop (one primary pump and one intermediate pump) have their pony motors furnished with power from the same buss. Thus, while it may be postulated that there may be many combinations of operable primary and intermediate pony motors, there are no common cause failure which could provide a mechanism for this. Nevertheless, a case has been analyzed which assumed the following: following a plant trip, the primary pump pony motors in loops 1 and 3 fail (loop 2 primary pump pony motor is available) and intermediate pump pony motor is not operating). The peak power to flow ratio seen was <0.9 (at 120 seconds into the transient). This event would be considered beyond the design basis because it induces three independent failures.

"Auxiliary Feedwater (AFW)
Both Diesel Driven (100%)
One Diesel Driven (50%)
One Turbine Driven (100%)."

The particular combination of AFW pumps that may be assumed will have no impact on the short term undercooling of the core. While the motor driven AFW pumps are designated "half capacity", this is with respect to the worst case conditions of:

- a pipe break on loop #1 with flow limited by the control valves,
- steam drum venting on loop #2, and
- superheater venting on loop #3.

If there are no leaks in the SG systems and normal venting takes place, either of the motor driven AFW pumps will furnish sufficient water to maintain steam drum levels. Thus, multiple failures are required to result in loss of the SG system as a heat sink from a feedwater standpoint.

"Protected Air Cooled Condensers (PACCs)
Natural Draft, Louvers Closed (0, 1, 2 or 3 loops)
Natural Draft, Louvers Open (0, 1, 2 or 3 loops)
Fans On, Louvers Open (0, 1, 2 or 3 loops)."

The Protected Air Cooled Condensers for each of the loops consists of two units rated at 7.5 MW_t, each with its own fan and set of louvers. The fans for the two units on each loop are furnished power from the same division of 1-E power.

If it is assumed that the louvers remain closed and the fans do not operate, the heat loss is negligible (0.45 MW_t per loop).

If the louvers are assumed to open but the fans do not function, the heat removal is approximately 30% (4.5 MW_t) of rated.

The PACCs are intended for long term decay heat removal. In the event that one or more units are not functioning, more venting (and thus more feedwater) would be required. However, so long as the drums do not dry out, decay heat and sensible heat will be removed independent of PACC operation.

"Number of Loops Initially Operating
3-Loop Operation
2-Loop Operation."

NRC review of CRBRP on two-loop operation is not being requested by the applicant at this time.

Question CS760.41

What are the bases (or the plans for determining the bases) for setting the conditions at which the DND signal will alert the reactor operator? What will be the operator responses to the DND signal in conjunction with other plant parameters? If these responses have not been determined, what are the plans for formulating them?

Response

Plans for determining the bases for setting the conditions at which the DND signal will alert the reactor operator are dependent upon results to be obtained from the on-going Run Beyond Cladding Breach (RBCB) program. Fuel assemblies having indications of fuel exposure beyond a defined limit, are to be removed from the core. Development of this limit is dependent on the development of appropriate technology through the RBCB Program which will assure the benign nature of operation with limited fuel sodium contact.

The applicant is committed to removing all failed fuel at each scheduled refueling outage. If a failed fuel assembly is characterized by the DND during reactor operation as having fuel exposure beyond the defined limit, then an orderly reactor shutdown will be initiated for the purpose of removal of the fuel exposure to sodium at that shutdown, the other failed fuel in the reactor may also be replaced.

Procedures for operator action will be formulated based upon the results from the on-going RBCB program and will be finalized during the OL review. A generalized approach is discussed in QCS760.39.

Question CS760.45

What change will the increased plutonium content of the fuel have on the offsite doses due to failure in the EVTM?

Response

The radioactivity source term for analysis of fuel failure in the EVTM is based on the maximum design basis conditions during plant life, including the use of LWR recycle plutonium (see PSAR Section 12.1.3) which contains the largest plutonium content.

Question CS760.77

Section 4.4.2.7 of the CRBR PSAR presents a discussion of pressure drop calculations and experimental results at full flow conditions. Is there a similar analytical and experimental base for low flow conditions? If not, how are pressure drops for low flow conditions determined? If there is a similar data base for low flow conditions, please provide detailed information in the form of calculated and experimental pressure drop data including uncertainty factors?

Response

The experimental results and pressure drop correlations and uncertainties presented in Section 4.4.2.7 of the PSAR were based on data which generally cover a range of Reynolds numbers from ~20 to 30% of the design Reynolds numbers to 100% or greater. In addition, the fuel, inner blanket and radial blanket rod bundle friction factors were based on preliminary data which range from less than 1% to greater than 100% of the design Reynolds numbers. The fuel assembly friction factor used in the PSAR is shown in Figure QCS760.77-1. It is significant that the fuel assembly rod bundle friction factor, the largest single primary system hydraulic resistance component was characterized over the full range of reactor flow rates. The remaining high resistance components are primarily orifice form losses which are not as Reynolds-number-sensitive as the friction losses. Consequently, low flow rate calculations performed with the PSAR correlations are valid down to approximately 20 to 30% flow and are expected to be quite close down to 1% flow.

Additional data and correlations such as those shown in Figures QCS760.77-2 and -3 for the blanket and control assembly rod bundle friction factors are under development. Also, a more recent fuel assembly friction factor correlation than the preliminary correlation shown in Figure QCS760.77-1 was developed based on 266 additional data points (Reference QCS760.77-1) but which differs from the Figure QCS760.77-1 correlation by only 0.2% at high flow rates and a maximum of ~2% in the transition region.

Data are becoming available on the overall fuel, blanket and control assembly pressure drops similar to those shown in Figure QCS760.77-4. Correlations to those data are under development. Flow tests to characterize orifices down to ~2% flow have been completed for the fuel and control assemblies. Similar testing of the radial blanket orifices located in the LIM is in progress. Orifice testing is under development for the inner and radial blanket assembly orifices down to low flow rates. The remaining hydraulic resistance components in the reactor inlet and outlet regions are all low pressure drop components which have been tested at high flow rates and will be extrapolated to low flow rates with an appropriate increase in uncertainty.

Rigorous data-based hydraulic correlations and uncertainties are being developed for all major reactor pressure drop components over the full range of operating flow rates. Final results will be presented in the FSAR.

Reference

QCS760.77-1 D. R. Spencer, R. A. Markley, "Friction Factor Correlation for 217-Pin Wire Wrap Spaced LMFBR Fuel Assemblies", ANS Transactions, 39, pp. 1014-1015, November, 1981.

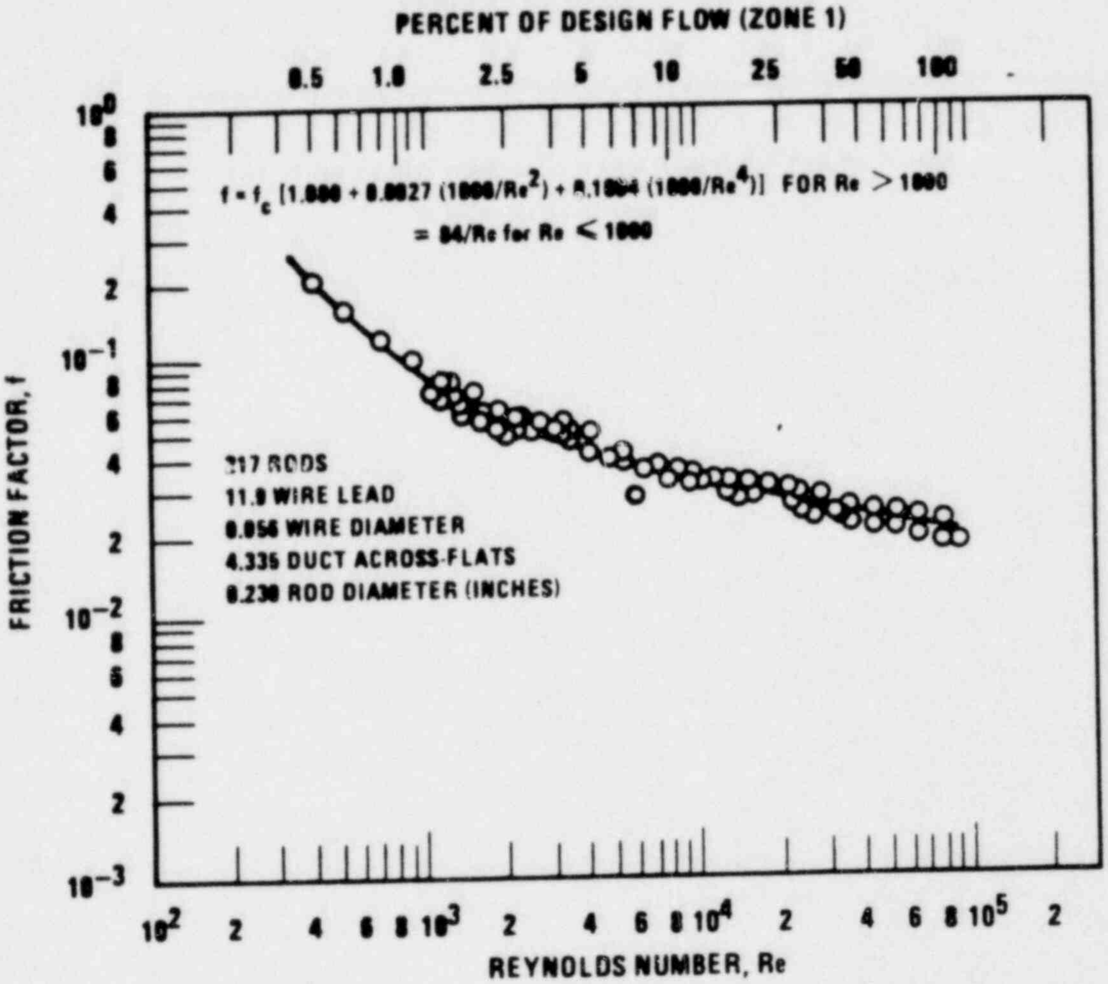

Figure QCS 760.77-1. Friction Factor Data and Correlation for 217 Pin Wire Wrap Spaced Fuel Assembly

7209-1

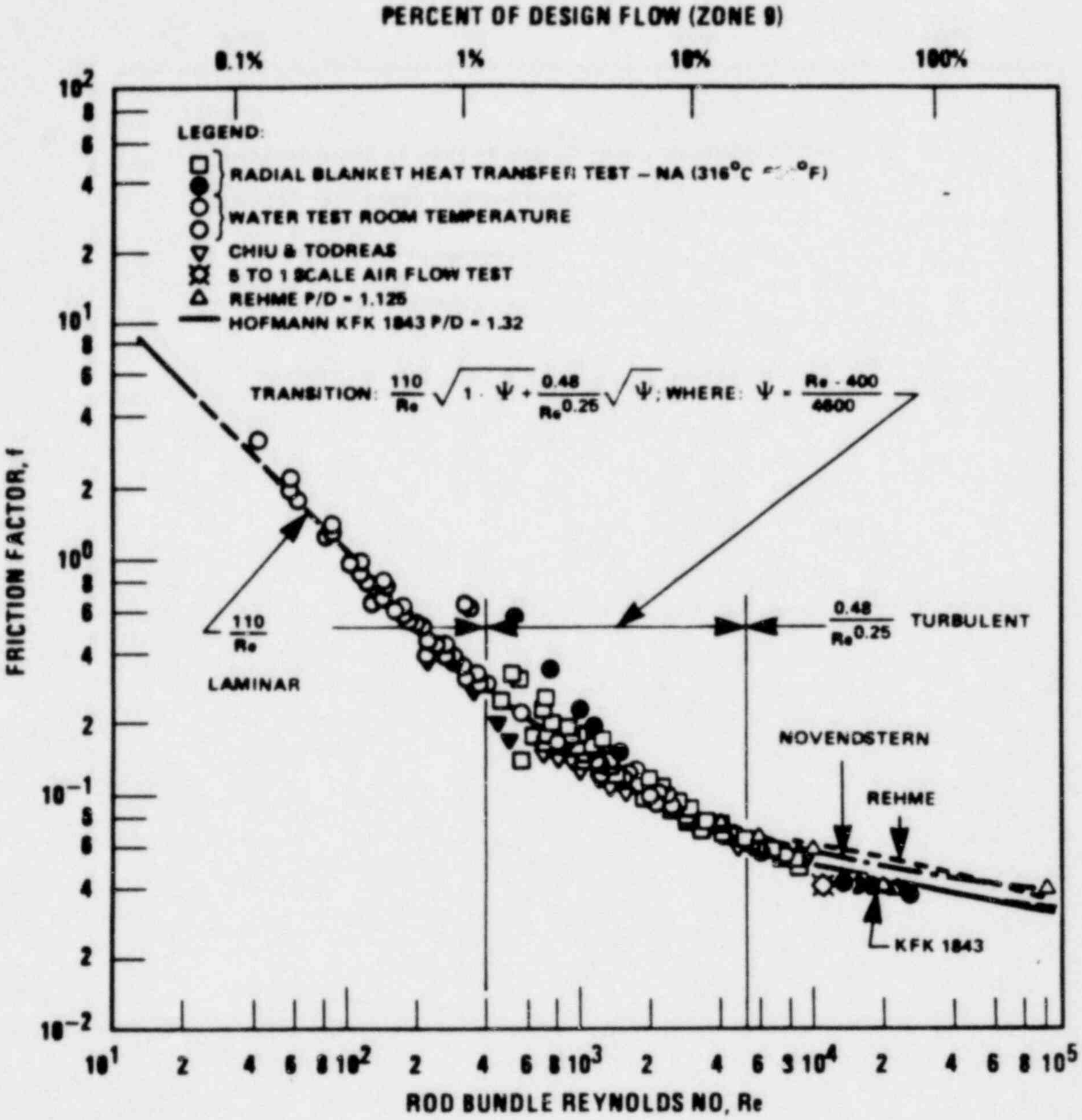

Figure QCS 760.77-2. Friction Factor Test Data for Tight Pitch to Diameter Rod Bundles With 4 Inch Wire Wrap Spacer Lead

7209-4

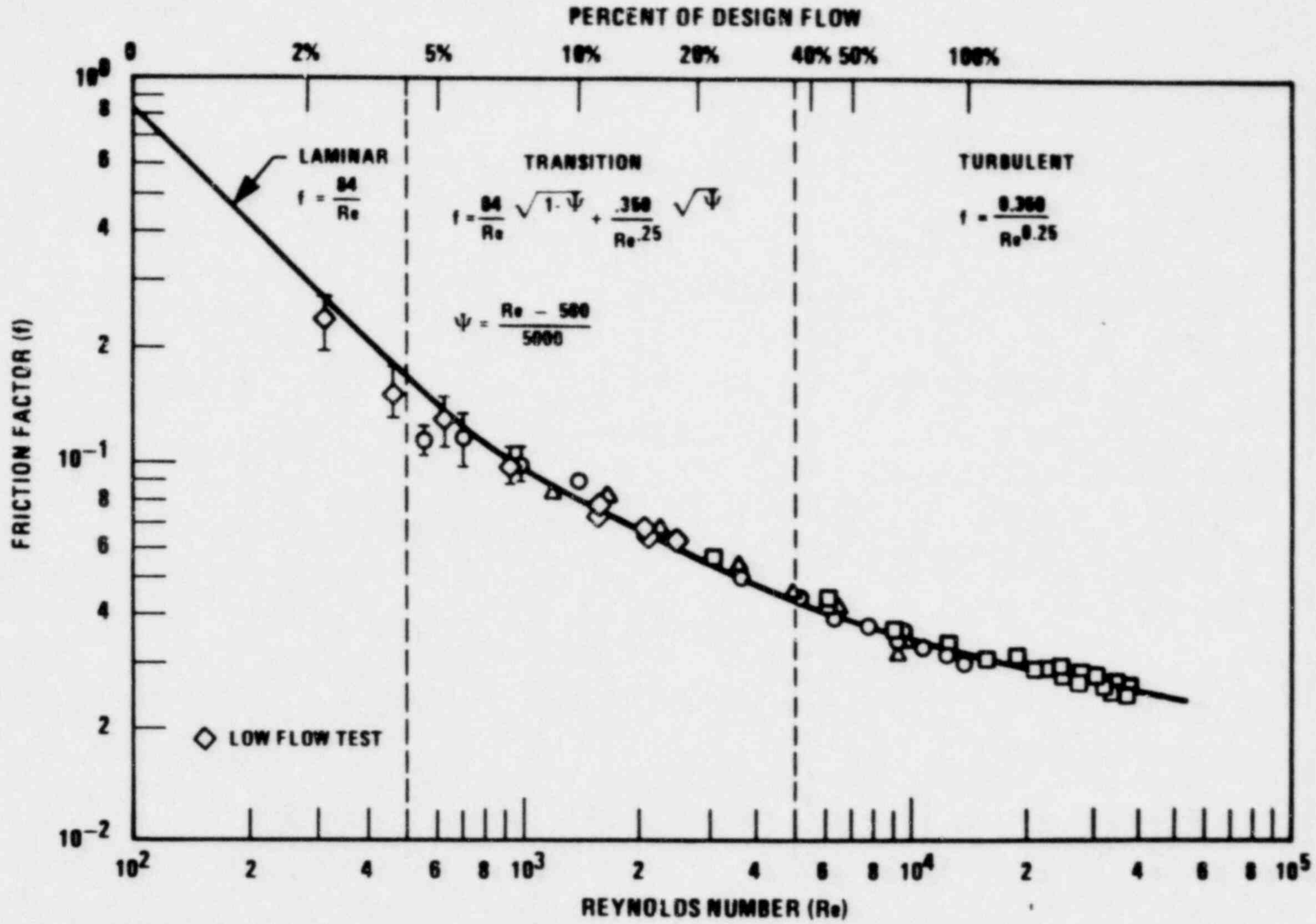

Figure QCS 760.77-3. Primary Control Assembly Rod Bundle Friction Factor

7209-3

QCS760.77-6

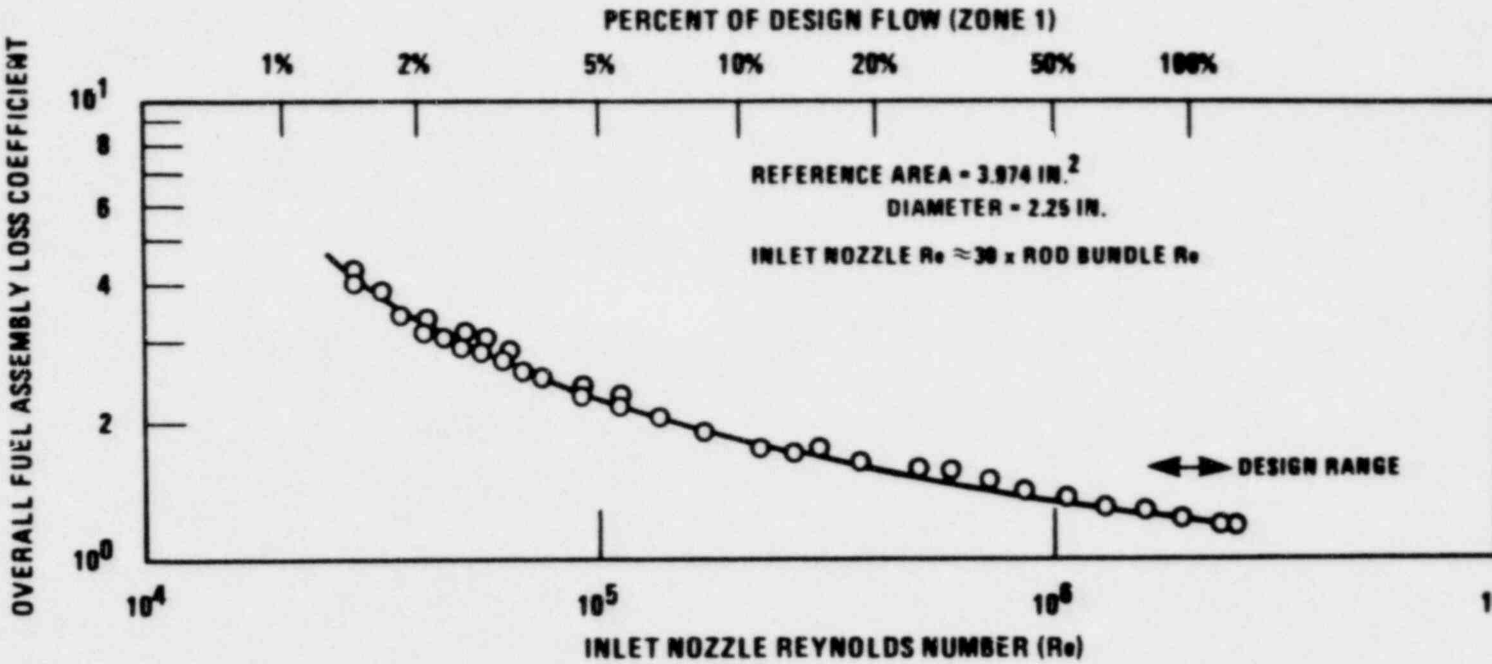

Figure QCS 760.77-4. Overall Fuel Assembly Loss Coefficient as a Function of Reynold Number from CRBRP Fuel Assembly Flow and Vibration Test

Question CS760.79

Do the uncertainty factors presented in Tables 4.4-18A through 4.4-31 apply to low flow conditions? If not, have uncertainty factors been determined for low flow conditions? If the aforementioned uncertainty factors do not apply to low flow conditions and none have been determined for low flow conditions, are there plans to perform experiments and calculations to determine these uncertainty factors? If the above referenced uncertainty factors do apply to low flow conditions, please provide data in the form of experimental and calculated results to support their use for low flow conditions.

Response

A) The uncertainty factors presented in Tables 4.4-18A through 4.4-31 were developed using data for high flow rates. Hydraulic uncertainties at low flows are discussed in the response to Question CS760.77.

B) Maximum (highest) core hot rod temperatures are calculated for the low flow conditions of natural circulation. These are used for design and safety evaluations. Application of full power and flow uncertainties is conservative since:

- At the low power flows and conditions of natural circulation, the effect of uncertainties which locally affect rod temperatures is insignificant since heat fluxes are small, i.e., local temperature differences between fuel, clad and coolant are minor for a particular axial position.

- Maximum cladding/coolant temperatures for natural circulation occur near the top of the heated core axial position and are thus the integrated effect of coolant over channel length. Uncertainties which affect integrated coolant temperature (enthalpy rise) are for full power and flow. At low flow, these uncertainties would be smaller for maximum temperature channels because of significant increse in inter- and intra-assembly flow and heat redistribution which results in a smaller temperature consequence due to an uncertainty in either power or flow.

C) for natural circulation analyses the full power uncertainties are increased multiplicatively for 3σ decay heat uncertainties. After the initial cooldown phase and during the period when maximum temperatures are reached the hot rod power is essentially all from decay heat, thus

$$P_{HC} = P \times (1 + \delta P') \times (1 + \delta P'')$$

where

P_{HC} = hot rod power generation from decay heat;

P = nominal decay power calculated for hot rod (includes radial and axial peaking);

$\delta P'$ = total full power uncertainty;

$\delta P''$ = 3σ decay heat uncertainty from ENDF-B4 data files

D) Numerous other uncertainties/conservative assumptions in addition to the ones previously mentioned are incorporated into all transient evaluations. These are summarized in Table QCS760.79-1. Conservatively all the aforementioned assumptions/uncertainties are assumed to occur simultaneously.

E) A prime example of the conservatism in this approach of maintaining full power/flow uncertainties for low flow plus the other assumptions of Item D is the comparison of calculated predictions using this method as compared to prototypic FFTF natural circulation experimental data shown by Figure QCS490.38-3. The second curve from the top shows 3σ hot channel calculations inforporating inter- and intra-assembly heat and flow redistribution effects; the lowest curve shows the measured temperature for the highest temperature core rod. The large difference between the top and bottom curves demonstrates the extreme conservatism of the overall uncertainty approach.

TABLE QCS760.79-1

MAJOR ASSUMPTIONS USED IN
TRANSIENT HOT ROD ANALYSIS

- Conservative plant THDV initial conditions (e.g., 750° reactor inlet)
- Worst case Doppler coefficient including uncertainties
- Minimum control rod shutdown worth (one stuck rod)
- Highest power and temperature hot rods at worst time in life
- Worst end of uncertainty range used for properties (e.g., fuel C_p) and fuel/clad gap conductance for both power and temperature calculations
- Maximum decay heat loads including time in life effects
- No credit taken for inter- and intra-assembly flow and heat redistribution
- Negative reactivity feedbacks neglected (e.g., core radial expansion, bowing, axial expansion of fuel and cladding)
- Conservative 0.2 second delay used for PPS logic, scram breaker and the control rod unlatch time delays

Question CS760.98

In Section 5.5.2.3.3 it is stated that the pipes were sized to limit mass flow rates to 20 feet per second for water through 175 fps for superheated steam. What are the maximum anticipated mass flow rates for steady-state operation and also during the various postulated events?

Response

Section 5.5.2.3.3 states that the steam and water piping is sized to limit fluid velocities at 100% power to the values indicated in the question. The fluid velocities at 100% power are given in the following table:

Piping Section	Fluid	Pipe Size, in.	Velocity, fps
Feedwater Inlet to Steam Drum	Water	10	17
Drum to Recirculation Pump	Water	18	11
Pump to Tee	Water	12	24
Tee to Evaporator Inlet	Water	10	17
Evaporator Outlet to Drum	Water/Steam	16	32
Drum to Superheater Inlet	Saturated Steam	12	104
Superheater Outlet to Isolation Valve	Superheated Steam	16	164
Drum Drain Line	Water	6	5

The fluid velocity design limits are based primarily on acceptable pressure losses in the system. Thus, the 24 fps velocity in the 12-inch pump-to-tee line was judged to be acceptable, since the pressure drop calculated for this section of line is only 5 psi, including the tee and 12 x 10 reducer. In addition, the standard outlet nozzle size for the selected recirculation pump was 12 inches.

The various postulated events (Table 5.7-1 of the PSAR) begin from initial conditions of 115% power. At this power level, the fluid velocity in the feedwater inlet, evaporator outlet, superheater inlet, and superheater outlet lines, will increase by 15% from the above values. The fluid velocities in the other lines will remain essentially constant.

The postulated events where line rupture is involved would have maximum anticipated fluid velocities dependent upon the nature of the rupture, but plant safety is maintained by isolation of the ruptured line section while other system components function to a safe shutdown situation.

PSAR Section 5.5.2.3.3 has been updated to reflect average water velocities not to exceed 25 fps.

4. Piping shall be designed with suitable access to permit in-service testing and inspection.

5. All "horizontal" piping shall be sloped. Steam traps and drain valves shall be located at the low points to permit complete draining of the piping.

6. Piping sizes shall be chosen such that average fluid velocities at the 100% plant power condition will not exceed the following values:

   a. water 25 fps

   b. water-steam mixture 50 fps

   c. saturated steam 125 fps

   d. superheated steam 175 fps

System Description

All Steam Generation System piping is shown in Figure 5.1-4. The design characteristics and ASME Code classifications are presented in Table 5.5-7.

The only field run piping planned for the steam generator system is non-safety class piping. The internal diameter of the piping will be 2 inches or less and is used for drain lines from steam traps. The design pressure would not exceed 100 psia and the design temperature would be less than 300°F.

The Seismic Category I design requirements are placed on the Steam Generation System's steam-water piping. Superheater and evaporator modules and the steam drum are provided with quick acting isolation valves. Design pressures of all piping are nominally 110% of the operating pressure at rated power.

The use and location of rigid-type supports, variable or constant spring-type supports, and anchors or guides will be determined by flexibility and stress analysis. Piping support elements will be as recommended by the manufacturers and will meet applicable code requirements. Direct weldment to thin wall piping will be avoided where possible.

Attachment and penetrations shall be designed and fabricated according to the ASME Code requirements.

Question CS760.134

The design basis assumes five (5) natural circulation occurrences per lifetime but the fuel damage analysis assumes only one.

Does this imply that a new fuel loading will be required after any natural circulation event?

Response

The CRBRP plant duty cycle is given in Appendix B, Table B-1, of the PSAR. For an emergency event like the natural circulation event, five (5) such events are considered per plant lifetime (one every 6 years) plus two consecutive occurrences of the most severe type (giving potentially seven total emergency events). For replaceable components such as the fuel assembly, one event of this type during the assembly lifetime is conservatively assumed for fuel design damage analysis.* However, after such an event, an assessment of core damage (including analytical methods) of course must be made before resumption of normal operation. If it were determined that there was little fuel rod damage and that acceptable fuel rod lifetime remained such that another event of this type could be taken, a new fuel loading would not be required.

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*Assuming a linear distribution of emergency events over the 30 year plant life would result in a 7/15 probability of occurrence over the 2 year fuel lifetime. This frequency fraction has been rounded-off to a single occurrence.

Question CS760.137

Please provide a list of the instrumentation and their functional requirements which are necessary to monitor and control the heat transport systems. Emphasize those instruments that the operator will use for decay heat removal under both normal and emergency conditions.

Response

A. Tables QCS760.137-1 through 3 provide the instrument list for the Heat Transport System (HTS) which may be used to monitor and control the systems.

B. The instruments the operator may use for decay heat removal under both normal and emergency conditions are identified below:

- o Primary Heat Transport System (PHTS) Temperature Hot Leg
- o Intermediate Heat Exchanger (IHX) Outlet Temperature (PHTS Cold Leg)
- o PHTS Pony Motor Speed
- o PHTS Flow
- o Intermediate Heat Transport System (IHTS) Temperature Hot Leg
- o Evaporator Outlet Temperature (IHTS Cold Leg)
- o IHTS Pony Motor Speed
- o IHTS Flow

All the instruments identified as being used by the operator for monitoring decay heat removal are provided on the Main Control Panel (MCP).

TABLE QCS760.137-1

PHTS INSTRUMENT CHANNELS

FUNCTION	SENSOR TYPE	MEASUREMENT USE	NUMBER OF SENSORS	RANGE
Pump Outlet Pressure	Nak Capillary	Surveillance Performance Evaluation PHTS/IHTS ΔP	2/Loop	0-200 psi
Reactor Inlet Press	Nak Capillary	Surveillance Performance Evaluation	2/Loop	0-200 psi
Pump Outlet Temp	RTD	Surveillance Performance Evaluation Calorimetric	2 dual/ Loop	300-1200°F
Reactor Inlet Temp	RTD	Surveillance Performance Evaluation Calorimetric Loop to Loop ΔT	2 dual/ Loop	300-1200°F
Main Loop Flow Evaluation	PM Flowmeter	Surveillance Performance	1/Loop	0-40000 gpm
IHX Outlet Temp	CA-T/C	Control, PPS	3/Loop	300-1000°F

TABLE QCS760.137-2

IHTS INSTRUMENT CHANNELS

FUNCTION	SENSOR TYPE	MEASUREMENT USE	NUMBER OF SENSORS	RANGE
IHX Inlet Temp	RTD	Surveillance Calorimetric	1 Dual/ Loop	300-800°F
IHX Outlet Temp	RTD	Surveillance Calorimetric	1 Dual/ Loop	300-1000°F
IHX Outlet Press	Nak Capillary	Surveillance	1/Loop	0-275 psig
Pump Inlet Press	Nak Capillary	Surveillance	1/Loop	0-150 psig
Pump Outlet Press	Nak Capillary	Surveillance	1/Loop	0-350 psig
Main Loop Flow	PM Flowmeter	PPS, Control	1/Loop	(1) 4000-40000 gpm
Evap. Outlet Temp.	CA-T/C	PPS	3/Loop	300-800°F

(1) PPS operating range. Functional range and indication are from zero.

TABLE QCS760.137-3

PHTS MAIN SODIUM PUMP INSTRUMENTATION

FUNCTION	SENSOR TYPE	MEASUREMENT USE	NUMBER OF SENSORS	RANGE
Pump Shaft Speed	Magnetic	Performance Surveillance	2/pump	0-1200rpm
Pony Motor Speed	Speed	Surveillance Indicator	1/pump	0-120rpm

IHTS MAIN SODIUM PUMP INSTRUMENTATION

FUNCTION	SENSOR TYPE	MEASUREMENT USE	NUMBER OF SENSORS	RANGE
Pump Speed	Magnetic	PPS, Control	5/Pump	(1)120-120rpm
Pony Motor Speed	Speed Indicator	Surveillance	1/Pump	0-120rpm

(1) PPS operating range. Functional range and indication are from zero.

Question CS760.138

Please include the core thermal response for item h (Section 5.7.3 - loss of one primary pony motor with check valve failure).

Response

This particular transient was analyzed using minimum decay heat along with various other conditions which were included to make the transient conservative from a plant piping and component design thermal transient standpoint. As such, the core thermal response from the analysis which produced the pump inlet transient for item h shows very modest temperatures. The core flows are, however, sufficiently high to keep the power to flow ratio well below 1.0. For example, at the time the flow reverses (100 seconds into the transient) in the loop with the failed pony motor (and a stuck open check valve), the core flow is 8.6%. At 600 seconds into the event, the reverse flow in the loop with the failed pony motor has steadied out at 200 lb/sec (5.2% of initial steady state loop flow) and the core flow is 6.9% of its initial flow. Thus the core temperatures will be less than their initial steady state temperatures.