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DRAFT

HFBR RESTART ACTIVITY A2.6:

REVIEW OF FSAR AND 60 MW ADDENDUM TO ASSURE CONSISTENCY OF OPERATION AT 40 MW

February 16, 1990
(Revised February 26, 1990)

Prepared for:

Mr. P. R. Shire
Los Alamos National Laboratory
N-Division, MS K557
P. O. Box 1663
Los Alamos, NM 87545

Prepared by:

Dasari V. Rao, Steven B. Ross, John L. Darby and Robert A. Clark
Science and Engineering Associates, Inc.
SEA Plaza
6100 Uptown Blvd. NE, Suite 700
Albuquerque, NM 87110

DISCLAIMER

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I. EXECUTIVE SUMMARY

The purpose of this task (HFBR Restart Activity A2.6) is to perform a review of the design basis accident (DBA) analyses sections of the 1964 HFBR-Final Safety Analysis Report; Volumes I and II, and the 1982 Addendum to the HFBR-FSAR for 60 MW operation to assure that operation at 40 MW will be consistent with these analyses. Additional documents utilized in the review included the Level 1 PRA for HFBR, HFBR-PDMs and HFBR-OPMs.

The review indicates that the 1964 FSAR-DBA analysis in incomplete in the sense that it did not analyze some of the important initiators for 1-loop operation that include:

- accidental throttling of primary flow control valves
- seizure of primary pump
- loss of secondary pump
- accidental throttling of secondary flow control valves
- rupture of secondary piping.

The first three initiators were later studied in the 1982 addendum. The other two initiators have not been examined to-date for 1-loop operation. It is recommended that the impact of these initiators be assessed prior to the restart, if 1-loop operation is chosen for the restart.

The review demonstrated that at 40 MW operation there are only a few accident initiators that will culminate in core damage (fuel melting and/or cladding failure) regardless of the availability of mitigating systems. For 1-loop operation these accidents include: fuel channel blockage, primary pump seizure, and large-large LOCA (a LOCA with effective break diameter > 2.81" is referred to as a large-large LOCA in this document as well as in PRA). Although all these accidents listed above could lead to core damage for 2-loop operation as well, the probability is expected be very low. Additionally for 1-loop operation, fuel damage may also be possible for three other initiators that lead to total loss of heat sink. These are: accidental throttling of secondary flow control valve (HCV-301A or HCV-301B), loss of both secondary pumps, and rupture of the secondary piping. The extent of core damage caused by these accidents could not be assessed since these cases were not analyzed for 1-loop operation. For 2-loop operation, however, the extent of core damage was shown to be minimal primarily due to low probability of occurrence. In view of this, it is recommended that 2-loop operation be considered for the restart. Additional advantages of the 2-loop operation include minimization of operator confusion and streamlining of operator response during various accident scenarios.

In all instances (1-loop or 2-loop 40 MW operation), the core damage was characterized by isolated cladding failure and/or localized fuel element melting. The secondary effects of the core damage, fuel element bowing and channel blockage that prevent natural circulation in the later stages, are expected to be negligible at 40 MW. Consequently, the effect of core damage on the primary systems/components performance can be assumed to be minimal.

II. SCOPE OF THE REVIEW

The purpose of this task (HFBR Restart Activity A2.6) is to perform a review of the design basis accident analyses sections of HFBR-Final Safety Analysis Report; Volumes I and II, and the Addendum to the HFBR-FSAR for 60 MW operation to assure that operation at 40 MW will be consistent with these analyses. This review included the following activities:

- Compilation of a list of design basis accidents analyzed and their impact on the safety systems and the fuel element integrity corresponding to single-loop 40 MW operation and two-loop 60 MW operation, and

- Comparison of these lists with each other to locate any inconsistencies associated with operation at 40 MW as well as to suggest a preferred mode of operation during low power restart.

II. METHODOLOGY

Accomplishment of the objectives stated above involved reviewing the following documents:

- HFBR Final Safety Analysis Report (FSAR); Volumes I and II, 1964
- Addendum to the HFBR-FSAR, 1978
- Addendum to the HFBR-FSAR for 60 MW operation, 1982
- Level I Internal Event PRA for the HFBR; Volume I: Summary and Results
- Level II Internal Event PRA for the HFBR; Volume II: Detailed Analysis and Appendices

Additional documents utilized in this task include:

- HFBR Plant Description Manual
- HFBR Technical Specifications and References
- HFBR Operations Procedures Manual, and
- HFBR Systems Descriptions and Systems List, SEA Report No. 89-258-23-A:1

This review was instrumental in the compilation of a set of tables that list DBAs analyzed and their impact on the safety systems and fuel elements for both 1-loop 40 MW operation and 2-loop 60 MW operation. These tables were then compared and contrasted with each other and also with the Level 1 PRA to examine the adequacy of the accident analyses presented in 1964 FSAR and 1982 Addendum. Based on this comparison we have made a few recommendations that were perceived to reduce the likelihood of core damage. The results of the review are presented in the following section.

IV. DETAIL

The details of the review are presented in the following two sub-sections. The first of these two sub-sections presents listing of the design basis accidents (DBA) analyzed and impact of each of the accidents for both single-loop and two-loop operation. The later subsection examines the consistency of operation at 40 MW with the DBA.

IV.1. Listing and Impact of DBA Analyzed in FSARs

Both the 1964 FSAR and 1982 Addendum for 60 MW operation were reviewed to compile a listing of design basis accidents analyzed in those documents, and their impact on the frontline and safety systems performance as well as the fuel element integrity. This review revealed that the 1964 FSAR-DBA analysis is incomplete, in that it did not analyze some of the important initiators. These initiators include: loss of power to the primary pump, accidental throttling of primary flow control valve, seizure of primary pump, and loss of secondary pump. These initiators were later analyzed in the 1982 addendum which examined the impact of these initiators for both two-loop 60 MW operation and single-loop 40 MW operation. Additionally there are two more initiators, (1) accidental throttling of secondary flow control valves and (2) rupture of secondary piping, that were not analyzed in either the 1964 FSAR or the 1982 Addendum. The importance of these initiators lies in the fact that both of these initiators lead to total loss-of-heat-sink resulting in core damage. The Level 1 PRA, however, studied the impact of these initiators in detail for 2-loop 60 MW operation, and briefly for 2-loop 40 MW operation. To the best of our knowledge, the impact of these initiators was not examined for 1-loop operation. Further discussions concerning this matter are presented in the later parts of this section in relevance to consistency of operation at 40 MW.

The results of the review of the 1964 FSAR and the 1982 Addendum are divided and presented in Tables 1 through 4. Tables 1 and 3 present the listing of DBAs analyzed for two-loop 60 MW operation and single-loop 40 MW operation, respectively. Also presented in these tables are the consequences of each of these initiators as pertinent to the fuel element integrity and radiation release. Tables 2 and 4 delineate the impact of these initiators on the frontline and safety systems performance and their availability. While compiling these tables particular emphasis was given to the determination of systems required for preferred shutdown and those that act as a first level backup. Additionally, it should also be pointed out that all of the tables mentioned above are based on conservative estimates of the system performance.

As pointed out in Tables 1 and 3 most of the DBAs analyzed pose no serious challenge to the fuel element integrity. There are, however, five exceptions in case of 60 MW operation where consequences of the initiators may include fuel melting and/or cladding failure. In all of these cases, the off-site dose consequences were within both the DOE and NRC guidelines for nuclear facilities. For 40 MW operation, low power densities combined with no requirement for forced flow through the core resulted in fewer cases of core damage. The DBA analyses results reveal that for 1-loop 40 MW there are only three initiators that might lead to core damage. These are: flow channel blockage, primary pump seizure and LOCA. In all instances that led to core damage at 40 MW operation, the damage was characterized by isolated cladding failure and/or localized fuel element melting. The secondary effects of the core damage that include fuel element bowing and channel blockage that prevent natural circulation in the later stages are expected to be negligible at 40 MW. Consequently, the effect of core damage on the primary systems/components performance can be assumed to be minimal. Tables 3 and 4 are based on this reasoning. Similar to 60 MW operation the off-site dose rates corresponding to these accident situation are expected to be minimal and bounded by DOE guidelines.

Table 1. List of DBAs analysed in 1982 Addendum and their impact for 2-loop 60 MW operation

Initiator #	*DBA Analysis Event Sequence*	*Impact of the Initiator for 2-Loop 60 MW operation*
1.0	Loss of Commercial Power:	Power outage of short duration, typically 1-2 secs, was listed as frequent. Long duration power shortage was not encountered until now. The impact of this initiator on various systems is as shown in Table 1-2. In case of power loss of short duration it is possible to resume full power operation upon retrieval of all primary and secondary systems. In all cases **the probability and frequency of core damage is very small.**
1.1	Short Duration (<3 sec)	
1.2	Prolonged Duration (>3 sec)	
2.0	Error in Placement of Fuel Element during Refueling	The event has *little or no impact on the front-line and support safety systems.* Scram is by manual action if the error is detected during operation. Usual shutdown procedures can be followed thereafter. **No core damage.**
3.0	Accidental Depressurization due to Opening of HCe-102	The event may *disrupt primary pump operation* by depressurizing the primary below NPSH. The minimum CHFR for fresh fuel core with conservative initial conditions is 2.71. **No core melting or cladding failure.**
4.0	Loss of Secondary Coolant Pump	*Loss of one primary cooler* is assumed. Scram by setback if the failed pump can't be replaced by 1 of the 4 available pumps. The CHFR calculations as well as pipe stress calculations reveal **no damage to fuel elements.**
5.0	Uncontrolled Rod Withdrawl	*No impact on front-line or support safety systems.* Even for the most conservative initial conditions, the minimum CHFR was found to be 2.78, which is much larger than the fuel damage limit (2.1). **No fuel damage.**
6.0	Accidental Throtteling of Primary Flow Control Valves	*Primary flow is assumed to be lost, irreversible.* A minimum CHFR of 2.23 is expected if both valves are throttled simultaneously. No **expected fuel damage or cladding failure.**
7.0	Fuel Channel Blockage	*No impact on safety systems.* Scram due to high D2O activity. **Cladding failure and fuel damage are probable.** The fuel melt is expected to be local and the off-site dose rate below 10 CFR 100 guidelines.
8.0	Loss of Power to Primary Pumps	*Except for loss of both primary pumps, status quo as per design.* Scram caused by three different instrument set points. Minimum CHFR is much larger than 2.1, the fuel damage limit. **Fuel or clad failure not expected.**
9.0	Accidental Opening of Manual Relief Valve in P-118	Disruption of primary motor expected. Even for most conservative initial conditions, the minimum CHFR is larger than 2.1. So, **no fuel or clad failure is expected.**
10.0	Thimble Flooding Accident	Small scale leaks can be handled as events with no impact on safety systems' performance. **Large leaks may result in fuel damage.** Off-dose rates were shown to be less than 10 CFR 100 guidelines.
11.0	Auxiliary Rod Break Accident	The minimum CHFR falls below 2.1. **Fraction of core over heated is less than 5 X 10E-5. Cladding failure likely.** Dose consequences below DOE guidelines and 10 CFR 100 guidelines.
12.0	Seizure of Primary Pump	Minimum CHFR was estimated to be less than 1.2. **Fuel and clad failure likely. Fraction of core overheated is less than 5 X 10E-4.** Off-site dose rates below 10 CFR 100 guidelines.
13.0	Primary System Pipe Break	The effects of LOCA are dependent on the break size and location. The FSAR-DBA assumed the worst-case scenario to evaluate the consequences. Corresponding to this situation **limited core damage is possible**, but fission product release is expected to be minimal.

Table 2. List of Systems/Components Required by HFBR-FSAR DBA for 60MW 2-Loop Operation†

	REACTOR TRIP				PRIMARY PUMPS		PONY MOTORS		SHUTDOWN PUMP		SECONDARY COOLING				RV DEPRESSURIZATION				
Initiator No.	Auto Scram	Setback	Manual Scram	Poison Water	GA101A	GA101B	GA101A	GA101B	GA102A	GA102B	No. of Pumps Required	Gravity Feed	Cross Connection	Other Systems	HCE 102	P102A,B P300A,B	Steam Condensing	D_2O Makeup	Natural Circulation
1.1	YES	–	MAN	MAN	C	C	AUTO	MAN	MAN	AUTO	0	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
1.2	YES	–	MAN	MAN	D	D	AUTO	MAN	MAN	AUTO	0	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
2.0	◊	–	◊	◊	0	0	◊	◊	◊	◊	0	◊	◊	◊	◊	◊	◊	◊	◊
3.0	YES	–	MAN	MAN	B	B	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
4.0	–	YES	MAN	MAN	A	A	AUTO	MAN	MAN	AUTO	1 of 4	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
5.0	YES	–	MAN	MAN	A	A	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
6.0	YES	–	MAN	MAN	A	A	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
7.0*	YES	–	MAN	MAN	A	A	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
8.0	YES	–	MAN	MAN	D	D	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
9.0	YES	–	MAN	MAN	B	B	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
10.0*	YES	–	MAN	MAN	A	A	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
11.0*	YES	–	MAN	MAN	A	A	AUTO	MAN	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
12.0	YES	–	MAN	MAN	D	A	AUTO	FAIL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
13.0*	YES	–	MAN	MAN	B	B	COND	COND	COND	COND	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN

◊ = Usual shutdown procedure
* = Assumed that fuel melt is local and does not affect entire system performance
† = See next page for explanation of other symbols and list of initiators

Systems required for preferred shutdown
Systems that act as first level backup
Systems failed or conditionally operational due to the initiator

Table 2. List of Systems/Components Required by HFBR-FSAR DBA for 60 MW 2-loop Operation.

Initiator #	Initiator
1.1	Loss of Commercial Power. Short Duration
1.2	Loss of Commercial Power. Prolonged Duration
2.0	Error in Placement of Fuel Element During Refueling
3.0	Accidental Depressurization due to Opening of HCe-102
4.0	Loss of Secondary Coolant Pump
5.0	Uncontrolled Rod Withdrawl
6.0	Accidental Throttling of Primary Flow Control Valves
7.0	Fuel Channel Blockage
8.0	Loss of Power to Primary Pump
9.0	Accidental Opening of Manual Relief Valve in P-118
10.0	Thimble Flooding Accident
11.0	Auxiliary Rod Break Accident
12.0	Seizure of Primary Pump
13.0	Primary System Pipe Break

Primary Pump Status

A	Operational
B	Available, but conditional
C	Coastdown, but breakers closed
D	Not available due to pump failure or loss of commercial Power

Other Symbols

MAN	Manual
AUTO	Automatic
COND	Conditional
FAIL	Failed or Assumed Failed

Table 3. List of DBAs analyzes in 1964 FSAR and 1982 Addendum and their impact for 1-loop 40 MW operation

Initiator #	*DBA Analysis Event Sequence*	*Impact of the Initiator for 1-loop 40 MW Operation*
1.0	Loss of Commercial Power:	Power outage of short duration, typically 1-2 sec, was listed as frequent. Long duration power shortage was not encountered until now. The impact of these initiators on various systems is as shown in Table 4. In case of power loss of short duration it is possible to resume full power operation upon retrieval of all primary and secondary systems. In all cases **the probability and frequency of core damage is very small.**
1.1	• Short Duration (<3 sec)	
1.2	• Prolonged Duration (>3 sec)	
2.0	Equipment Failures:	Detection of this class of initiators is very difficult until fuel failure. Therefore, in all these cases, irrespective of proper functioning of on-line and support safety systems **fuel cladding failure is possible.** Even if it is assumed that all fission gasses from the failed fuel elements are released into the confinement building, the off-site dose consequences are much less than the 10 CFR 100 guidelines.
2.1	• Fuel Channel Blockage	
2.2	• Fuel Cladding Failure	
2.3	• Instrument system malfunction	
3.0	Loss of Pressure Accidents:	Events of this class eventually disrupt proper functioning of the primary and shutdown pumps, if no preventative measures were taken. In both these cases, the minimum CHFR is much larger than 2.1, the fuel damage limit. Consequently, these initiators are **not expected to result in fuel melting and/or cladding failure.** Automatic opening of flow reversal valves following loss of forced flow (primary and shutdown pump tripping) would enable decay heat removal.
3.1	• Accidental Depressurization due to Opening of HCe-102	
3.2	• Accidental Opening of Manual Relief Valve in P-118	
4.0	Reactivity Accidents:	Initiators of this class result in increased reactivity causing the reactor power to increase rapidly until the trip setting (48 MW) is reached and the autoscram action causes reactor shutdown. At higher powers (60 MW operation, for example) this rapid increase in power may challenge the fuel element integrity. At 40 MW, however, the impact of the initiator is acceptable, in that **it does not pose any threat to the fuel element integrity.** The estimated minimum CHFR is larger than the fuel damage limit (2.1).
4.1	• Uncontrolled Rod Withdrawl	
4.2	• Thimble Flooding Accident	
4.3	• Auxiliary Rod Break Accident	
4.4	• Light water contamination	
4.5	• Cold water accident	
5.0	Loss of Primary Pump:	All these initiators result in loss of primary pump, at least temporarily. Resultant low primary flow rates cause reactor scram. The shutdown pump can then be used for decay heat removal. The natural convection cooling of the core provides the first level of redundancy. Once again due to low operating power (40 MW), the impact of power loss to primary pumps and accidental throttling of flow control valves is not as severe as demonstrated by large CHFRs estimated for these transients. However, MCHFR estimates for the case of seizure of primary pump are less than 2.1 which indicates **possible clad failure and/or fuel melting.**
5.1	• Power Loss to Primary Pump	
5.2	• Accidental Throttling of Primary Flow Control Valve	
5.3	• Seizure of Primary Pump	
6.0	Primary System Rupture:	In all cases of ruptures the core remained covered and fuel melting was not expected. The major consequence of rupture in the primary system was found to be leakage of steam produced in the core to the exhaust gas system and stack. Heavy water could be drained from the rest of the system for makeup, but poisoned light water is preferable to limit tritium release. The off-site dose consequences of these initiators were found to be bound by the DOE guidelines as well as the NRC guidelines.
6.1	• Primary Pipe Break	
6.2	• Reactor Vessel Rupture	

Table 4. List of Systems/Components Required by HFBR-FSAR DBA for Single Loop 40 MW Operation†

	REACTOR TRIP				PRIMARY PUMP	SHUTDOWN PUMPS		SECONDARY COOLING				RV DEPRESSURIZATION				
Initiator No.	*Auto Scram*	*Setback*	*Manual Scram*	*Poison Water*	*GA101A*	*GA102A*	*GA102B*	*No. of Pumps Required*	*Gravity Feed*	*Cross Connection*	*Other Systems*	*HCE 102*	*P102A,B P300A,B*	*Steam Condensing*	*D2O Makeup*	*Natural Circulation*
1.1	YES	-	MAN	MAN	COAST DOWN; BUT BREAKERS ARE CLOSED	MAN	AUTO	0	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
1.2	YES	-	MAN	MAN	NOT AVAILABLE	MAN	AUTO	0	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
2.1	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
2.2	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
2.3	-	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
3.1	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
3.2	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
4.1	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
4.2	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
4.3	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
4.4	YES	-	MAN	MAN	OPERATIONAL	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
5.1	YES	-	MAN	MAN	COAST DOWN; BUT BREAKERS ARE CLOSED	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
5.2	YES	-	MAN	MAN	NOT AVAILABLE	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
5.3	YES	-	MAN	MAN	NOT AVAILABLE	MAN	AUTO	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN
6.0	YES	-	MAN	MAN	NOT AVAILABLE	COND.	COND.	1 of 5	MAN	MAN	MAN	AUTO	MAN	MAN	MAN	MAN

Systems required for preferred shutdown — Systems that act as first level backup — Systems failed or conditionally operational due to the initiator

† See Table 3 for the list of initiators

IV. 2. CONSISTENCY OF OPERATION AT 40 MW

The safety systems configuration and the level of automation have a large impact on the outcome of any accident. Therefore, these factors are very important in the determination of consistency of operation at 40 MW and hence are examined first.

From Tables 1 through 4 it is clear that the HFBR is equipped with various systems and components that are capable of mitigating a variety of accidents. These required systems and components are automated, in most cases. The systems and components that are controlled manually from the control room or the operations area provide the first or second level redundancy depending on the impact of the initiator on systems performance. Thus the operators actions provide a backup that is only required in case of failure of automated systems. The only exception to this is a LOCA in the primary system where the operator actions are essential to isolate the break and mitigate the accident. It can therefore be concluded that operation at 40 MW is consistent with the accident analyses and there are no major modifications suggested for such operation. There is, however, a recommendation relevant to the mode of operation at the restart. It is suggested that the restart activities be focused on 2-loop operation rather than 1-loop operation. The justification for this recommendation is provided in the following paragraphs.

IV.2.1 Preferred Mode of Operation[1]:

There are two possible modes of operation at 40 MW, 1-loop operation (only one primary loop is used for heat removal during steady state) and 2-loop operation (both primary loops are used for heat removal). The accident analyses sections of the 1964 FSAR and the 1982 Addendum analyzed design basis accidents for 1-loop 40 MW operation. The level 1 PRA document examined the impacts of some of the important initiators for 2-loop 40 MW operation. Table 5 compares the impact of various initiators corresponding to both cases. As evident from this table as well as from Tables 3 and 4, for operating power equal to or less than 40 MW most of the initiators pose no severe challenge to the fuel element

[1] The original intent of this subsection was to establish the relative merits of 2-loop 40 MW operation in comparison to 1-loop operation. However, we have been recently informed that BNL has already decided to operate with 2-loops. Nevertheless, we have decided to leave this section intact as it is the only section that provided a methodical comparison of the impacts of each initiator for 2-loop and 1-loop operation. Additionally, we foresee its usefulness in future when similar comparison is required.

integrity. Of the initiators examined in these documents, flow channel blockage, primary pump seizure and primary system rupture are, however, three exceptions where the consequences may include fuel melt and/or cladding failure due to the onset of CHF. The 2-loop operation reduces likelihood of such consequences considerably (see Table 5). As pointed out in Table 5, in the case of 2-loop operation fuel melt and/or cladding failure are only possible if both primary pumps (not just one as in the case of 1-loop operation) are seized simultaneously. Probability of such a coincidence is very small when compared to the probability of failure of a single pump. Similarly, 2-loop operation also reduces the likelihood of extensive damage to fuel elements during a large-large LOCA (a LOCA with effective break diameter larger than 2.81" is classified as large-large LOCA). As pointed out in Table 5, a large amount of liquid inventory associated with 2-loop operation together with the pony motor that operates until the liquid level falls below 177" enables forced flow through the core for a longer period, up to 40 secs. This reduces the amount of core damage. These two cases demonstrate quantitatively the clear advantage of 2-loop operation over the single loop operation. There are additional advantages of 2-loop operation that are pertinent to loss of heat sink accidents. This advantage can be demonstrated by considering some of the initiators related to the secondary cooling systems: accidental closure of the secondary flow control valve and rupture in the secondary piping. Neither of these initiators were analyzed in the DBA although they were examined in the level 1 PRA for 2-loop 60 MW and 40 MW operation. Due to lack of core damage frequency estimates for 1-loop operation, it is not possible to provide a quantitative comparison of the impact of these initiators for 1-loop and 2-loop operation. Consequently, the discussions in the following paragraphs are qualitative in nature.

Table 5. Comparison of the impact of each of the DBAs for 1-loop with those for 2-loop

DBA Analysis Event Sequence	*Impact of the Initiator for 1-loop 40 MW Operation*	*Projected Impact of the Initiator for 2-Loop 40 MW Operation*
Loss of Commercial Power: • Short Duration (<3 sec) • Prolonged Duration (>3 sec)	Consequences of this accident have been analyzed in the 1964 FSAR. The probability and frequency of core damage for these initiators was determined to be negligible.	Level 1 PRA analyzed this accident in detail. The probability and frequency of core damage is once again negligible, even smaller than that for 1-loop operation becuase of the availability of additional DC operated pumps: the pony motors.
Equipment Failures: • Fuel Channel Blockage • Fuel Cladding Failure • Instrument system malfunction	The mode of operation has little or no impact on the outcome of these accidents because the initiators are not primary cooling related. As pointed out in Table 4, these inititators may result in fuel melting and/or cladding failure.	
Loss of Pressure Accidents: • Accidental Depressurization due to Opening of HCe-102 • Accidental Opening of Manual Relief Valve in P-118	Event sequence progression for loss of pressure accidents is independent of the mode of operation: single-loop or two-loop. In both cases the final outcome is acceptable, in that no fuel melting and/or cladding failure are expected. In the case of two loop operation, however, presence of an additional pony motor increases the probability of sufficient forced flow through the core following the accident.	
Reactivity Accidents: • Uncontrolled Rod Withdrawl • Thimble Flooding Accident • Auxiliary Rod Break Accident	Once again, the mode of operation has little or no impact on the outcome of these accidents because the initiators are not primary cooling related. Becuase of the low operating power, rapid increase in the heat flux due to reactivity insertion does not cause the onset of CHF. MCHFR is larger than 2.1 indicating no expected clad failure.	
Loss of Primary Pump: • Power Loss to Primary Pumps • Accidental Throttling of Primary Flow Control Valve • Seizure of Primary Pump	If complete failure of primary pump is assumed, then one must rely on shutdown pumps and the opening of the flow reversal valves for decay heat removal. These redundant systems are adequate and no core damage is expected in the first two cases. Seizure of primary pump, however, could have serious consequences; the conservative analysis of this accident scenario indicates possible core damage due the to onset of CHF.	The advantage of two-loop operation is most prevelant in these cases. In case of two-loop operation the consequences of these accidents are severe only if both primary pumps fail simultaneously. Probability of such an ocurrence is minimal. Which demonstrates a clear advantage of 2-loop operation in comparison to the one-loop operation. Failure of a single loop due to any of the three initiators can be easily mitigated and poses no threat to the fuel element integrity.
Primary System Rupture: • Primary Pipe Break • Reactor Vessel Rupture	In all cases analysed, the core was reported to be covered with water at the end of the accident. During this midstage *for a large-large* LOCA (*effective break diameter* larger than 2.81") all forced flow is lost in less than 40 secs. Under these particular circumstances core damage is possible. Otherwise no core damage is anticipated. The only safety concern pertains to release of radioactive steam through the break.	Similar consequences are possible in 2-loop operations also. But there are some major differences that allow for accident mitigation. Large amount of *liquid* inventory together with pony motor operation (Trip level 177") enable forced flow through the core for a longer period, approximately 40 secs. This results in much less area of core to be damaged. The drawback associated with 2-loop operation is the fact that more piping length increases the probability of the ocurrence of a LOCA.

Accidental Throttling of Secondary Flow Control Valve

First we will study the consequences of accidental throttling of a secondary flow control valve (HCV-301A or HCV-301B) for 1-loop operation as well as 2-loop operation. Because, this initiator is very similar to the 'loss of both secondary pumps accident', the following discussions are expected to be applicable to the case of accidental failure of both secondary pumps. In the case of 2-loop operation, accidental closure of one of these valves results in a partial loss of heat sink, that is loss of one-of-two primary heat exchangers. Resultant increase in the primary coolant temperature causes reactor shutdown through setback. Onset of CHF is not expected and no fuel melting is expected. Once the reactor is shutdown, the decay heat can be easily removed by the operating heat exchanger. The shutdown heat exchanger and the shutdown loop provide a backup and steam condensing provides second level backup. For 1-loop operation, however, the consequences of this initiator are different. Because the heat removal during steady state is accomplished through a single primary heat exchanger, accidental closure of secondary flow control valve in the loop would lead to a total loss of heat sink. This action causes a rapid increase in the primary temperatures leading to reactor shutdown by setback. This rapid increase in coolant temperature causes increase in the core temperatures (fuel temperature, piping temperatures, reactor vessel temperature), and may also lead to the onset of CHF. As noted in the 1982 FSAR, this rapid increase in the core temperatures increases thermal stresses in the heat exchanger pipes, beam tubes and other piping, which in turn may lead to failure some of these components.

If it is assumed that fuel element integrity and the system integrity is not threatened during this accident (unproven as of now), then the mitigating steps available for the operator is tripping of the primary pumps so that shutdown heat exchanger can be used for decay heat removal and core cooling. The backup to this action would be natural convection together with steam condensing or last resort poison water addition. From the explanation presented above, it is clear that operator action is essential to avert serious consequences associated with this accident if 1-loop operation is chosen. In the case of 2-loop operation, such an operator intervention is necessary only if both control valves are simultaneously shutdown and/or both secondary pumps simultaneously trip; probability of such a coincidence is relatively low. Even in case of such a failure there is more time available for operator action in the case of 2-loop operation relative to 1-loop operation due to increased liquid inventory in the loop. These discussions clearly demonstrate the relative advantage of 2-loop operation, at least qualitatively. If it is felt, however, that this advantage is minimal, then a detailed safety analysis of this accident should be provided in support of 1-loop operation.

Secondary Coolant Pipe Rupture:

The other initiator under consideration, secondary coolant pipe rupturing, also results in total loss-of-heat-sink for 1-loop operation leading to a rapid increase in the primary coolant temperatures. The outcome of such an accident, based on conservative estimates, could be very similar to the previous case: fuel melt and/or cladding failure. While a similar situation (total loss-of-heat-sink) is also possible in the case of 2-loop operation, the probability is low. Even in the case of such an extreme situation the primary coolant temperature increase is expected to be moderate considering the large liquid inventory associated with 2-loop operation. This is another accident that needs to be analyzed and consequences quantified prior to restart, should 1-loop operation be chosen for operation.

Relevant Engineering Insight:

In spite of the above arguments, should 1-loop operation be chosen, the drawbacks can be overcome by instrumenting a new logic circuit that would automatically scram the reactor and trip the primary pumps when the secondary flow rate to the heat exchanger falls well below the steady state value. These automated actions would cause rapid decline in the heat flux due to scram as well as introduce a heat sink through the use of shutdown heat exchanger. Together, these measures would deter core overheating and preclude operator error. Although the modification is perceived by us to be necessary only for 1-loop operation, it is helpful for 2-loop operation as well.

Other Relevant Factors:

There are other advantages associated with 2-loop operation. These include unified operating procedures between 40 MW and 60 MW operation which would certainly reduce operator confusion. In this case, operator reactions and actions can be streamlined and mitigating choices can be minimized. Additionally, continuous operation with 2-loop, irrespective of the operating power, would simplify checking of the instrument set-points and trip points. Although these drawbacks might be overcome with extensive training, it is preferable that potential operator confusion be minimized.

In conclusion, although 1-loop operation is safe in absolute terms (low probability and frequency of core damage), in relative safety terms it is qualitatively less desirable than the 2-loop operation. It is, therefore, recommended that 2-loop operation be chosen for low power restart activity. We recognize that this choice is associated with certain delays in restart due to the necessity to rewrite the technical specifications for 40 MW 2-loop operation. However, if 1-loop operation is chosen then it is desirable to update the DBA analysis to quantify the impact of two additional initiators: (1) accidental throttling of secondary flow control valves, and (2) secondary system pipe rupture.

V. CONCLUSIONS

The review indicates that the 1964 FSAR-DBA analysis is incomplete in the sense that it did not analyze some of the important initiators for 1-loop operation that include:

- accidental throttling of primary flow control valves
- seizure of primary pump
- loss of secondary pump
- accidental throttling of secondary flow control valves
- rupture of secondary piping.

The first three initiators were later studied in the 1982 addendum. The other two initiators have not been examined to-date for 1-loop operation. It is recommended that the impact of these initiators be assessed prior to the restart, if the 1-loop operation is chosen for the restart.

The review demonstrated that there are only a few accident scenarios that culminate in the core damage (fuel melting and/or cladding failure). For 1-loop operation these accidents include: fuel channel blockage, primary pump seizure, and large-large LOCA regardless of the availability of frontline and safety systems. Although all these accidents listed above could also lead to core damage for 2-loop operation as well, the probability is expected be very low. Additionally for 1-loop operation fuel damage is also be possible for three other initiators that lead to total loss of heat sink. These are: accidental throttling of secondary flow control valve (HCV-301A or HCV-301B), loss of both secondary pumps, and rupture of the secondary piping. The extent of core damage caused by these accidents could not be assessed since these cases were not analyzed for 1-loop operation. In view of this, it is recommended that 2-loop operation be considered for the restart. Additional advantages of the 2-loop operation include minimization of operator confusion and streamlining of operator response during various accident scenarios.

In all instances (1-loop or 2-loop 40 MW operation), the core damage was characterized by isolated cladding failure and/or localized fuel element melting. The secondary effects of the core damage, fuel element bowing and channel blockage that prevent natural circulation in the later stages, are expected to be negligible at 40 MW. Consequently, the effect of core damage on the primary systems/components performance can be assumed to be minimal.