8005060 064 to DAC:ADM:
CENTRAL FILES
PDR:HQ
LPDR
TIC
~~NSIC~~
STATE

Public Service Company of Colorado

March 18, 1980
Fort St. Vrain
Unit No. 1
P-80051

Mr. Karl V. Seyfrit, Director
Nuclear Regulatory Commission
Region IV
Office of Inspection and Enforcement
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76012

Subject: Environmental Qualification of Class IE Equipment

References: IE Bulletin 79-01B
P-80037, March 4, 1980
Swart to Seyfrit

Dear Mr. Seyfrit:

As indicated in our letter of March 4, 1980 (P-80037), PSC is hereby submitting a partial response to IE-79-01B. Along with the submittal, we are summarizing the actions that are being taken as a result of IE Bulletin 79-01B. Justification is provided for not responding in areas that we feel do not apply to an HTGR. Also provided is the current status of our response and the schedule for its completion.

General

As previously submitted to the NRC (Attachment D to P-77137, Millen to Denise, June 15, 1977, enclosed as Attachment D), the double-ended rupture of a cold reheat pipe in the Reactor Building produces the most severe environmental conditions in that building. Similarly, the double-ended rupture of a hot reheat pipe produces the most severe environmental conditions in the Turbine Building. For the conditions resulting from a double-ended rupture of a hot or cold reheat pipe, components were qualified to the distance nearest the opposite loop steamline or to the nearest steamline if the component must function even if its own loop fails. If the distance was equal to or greater than 20 feet, the conditions of the 20 foot curve for the reactor or turbine buildings was used to qualify the components.

Partial Submittal (Attachment A)

Our submittal consists of a listing of Class IE electric equipment within the "accident zone" that is required to electrically function under accident conditions to provide safe shutdown cooling.

The submittal is formatted similarly to enclosures #1 and #2 of IE-79-01B.

Equipment locations are given in the column entitled "SRB LOCATION", which is defined in Attachment E.

The submittal that is similar to enclosure #2 is a tabulation of "tagged items." The term "tagged items" refers to equipment, instruments or components that are identified by a specific number. The numbers are alpha numeric in nature and provide the following information:

1. The alpha portion identifies the type of component, ie; as, P=Pump, HS=Hand Switch, etc. See Attachment B for a complete list of the Alpha prefixes.

2. The numeric portion identifies the system involved. Specifically, the first two digits identify the system involved. See Attachment B for a complete list of system numbers and names.

The submittal that is similar to enclosure #1 consists of "untagged" or "subtier" components. These subtier components are the many items such as relays, switches and other control components that are required as a part of the control loop or circuitry to make the tagged (equipment) item function. These types of items do not carry any specific identification other than the manufacturer's model or part number and are generally not shown on the plant P&I drawings.

Summary of Actions

The following areas are being investigated in relation to our response to IE-79-01B:

1. The environmental test records for the possible inclusion of additional subtier items.

2. The computer programming required to format our final response similar to enclosure #3 of IE-79-01B.

3. Equipment suppliers are being contacted regarding aging.

4. The Emergency Procedures are being reviewed.

Areas Not Applicable to an HTGR

Pressure:

The FSV HTGR does not have a containment building, therefore, there is no storage of blowdown steam and thus no ambient pressure buildup.

The reactor and turbine buildings are both vented. Therefore, pressure transients resulting from a high energy line break will be very localized and short-term in nature.

Further details about steamline rupture analysis at FSV may be found in Attachment D to P-77137, dated June 15, 1977.

Relative Humidity:

For the same reasons as discussed above under the pressure heading, Relative Humidity is not a problem at FSV after a high energy line break.

Chemical Spray:

No chemical sprays are utilized at FSV for cooling.

Radiation:

There are no radiological concerns directly associated with a high energy line break at FSV. That is, the process fluid (steam or feedwater) is not contaminated.

To postulate a radiological incident DBA #1 "Permanent Loss of Forced Circulation" and DBA #2 "Rapid Depressurization/Blowdown" were considered. DBA #1 provides the worst case radiological conditions, but the overall radiological concerns are minimal.

Complete details of this accident may be found in Section 2.1.6.b of P-79312 (Swart to Varga) dated December 28, 1979, enclosed as Attachment C.

In summary, the peak doses in the Reactor Building following DBA #1 are as follows:

Location	Peak Gamma Dose Rate	Time of Peak	180 Day Accumulated Dose (Rem)
Reactor Building (above Refueling Floor)	1.4 R/hr	24 hours	400

In conclusion, the reactor building will be accessible for short-term operations following such an accident. The accumulated doses indicated above would have no operational effect on the Reactor Building equipment.

Submergence

The nuclear reactor at Fort St. Vrain is cooled by gas and not water. Shutdown of the reactor is accomplished by control rod insertion. Emergency shutdown is accomplished by pressurized shutdown hoppers that drop boron balls into the reactor. Water is not used for shutdown or emergency core spray of the reactor in an HTGR. Venting of the reactor cooling quench water and/or primary coolant water into the containment sump is not applicable for Fort St. Vrain. Therefore, submergence is not deemed to be a problem at Fort St. Vrain.

Schedule

Component evaluation worksheets similar to IE-79-01B Enclosure 3 will be submitted in a preliminary form within approximately 2 weeks, along with any revisions to the master list.

After the above is completed, revised versions of the master list and component worksheets will be supplied on a weekly basis if revisions are required.

The response to this bulletin and the schedule for its completion is based upon the use of the firewater system as the source of motive power for the helium circulators and water to cool a steam generator as described in the FSV FSAR Section 14.4. This mode of reactor cooling utilizes only seismically and environmentally qualified equipment components and systems. With this in mind, there should be no problem in meeting the 90 day response deadline for IE-79-01B.

If the review of the FSV Emergency Procedures or any other item has a major impact on the schedule the staff will be so advised.

Very truly yours,

Frederic E. Swart

Frederic E. Swart, Manager
Nuclear Project Department

FES/MEN:pa

Attachments

ATTACHMENT A

POOR ORIGINAL

COMPONENT 1		SCI SUPPLIER 23	SCI MODEL 24	PRODUCT 4	SRB LOC 35
SUBT-	009	ALLEN BRADLEY	700N400	RELAY	RX2
SUBT-	010	ALLEN BRADLEY	700N800	RELAY	RX2
SUBT-	011	ASCO	HB8302C25F	VALVE	RX2
SUBT-	012	ASCO	HB8302C29U	VALVE	RX2
SUBT-	013	ASCO	HB8302C29F	VALVE	RX2
SUBT-	015	ASCO	LB8320A108	VALVE	TB2
SUBT-	016	ASCO	HB8302C25G	VALVE	RX2
SUBT-	017	ASCO	8302C26U	VALVE	TB2
SUBT-	018	ASCO	HB8302C25U	VALVE	RX2
SUBT-	019	ASCO	8320A89	VALVE	RX2
SUBT-	023	BARKSDALE	12453	VALVE	RX2
SUBT-	024	COLLINS	SS409	TRANSMITTER	RX2
SUBT-	071	GENERAL ELECTRI	CR120A04222AA	RELAY	RX2
SUBT-	072	GENERAL ELECTRI	CR2940UM200AC	CONTROL SW	RX2
SUBT-	073	ITT GEN CONT'L	AH91	MOTOR	RX2
SUBT-	076	KAHN COMPANIES	3784-508	THERMOSTAT	TB2
SUBT-	078	KAHN COMPANIES	3784-512	VALVE	TB2
SUBT-	079	KAHN COMPANIES	3784-514	VALVE	TB2
SUBT-	080	KAHN COMPANIES	3784-515	CONTACTOR	TB2
SUBT-	081	KAHN COMPANIES	3784-520	SWITCH	TB2
SUBT-	082	KAHN COMPANIES	3784-521	SWITCH	TB2
SUBT-	083	KAHN COMPANIES	3784-522	VALVE	TB2
SUBT-	093	KIELEY MUELLER	467	AIR SET	RX2
SUBT-	101	MASONEILAN	8005	POSITIONER	RX2

SUBT-	123	MOOG	72-101D	VALVE	TB2
SUBT-	129	PARKER HANNIFIN	D1W20HVY-10	VALVE	TB2
SUBT-	131	PARKER HANNIFIN	3MD20UBHP-38	VALVE	TB2
SUBT-	205	ASCO	8300C9U	VALVE	TB2
SUBT-	209	ASCO	8302C25F	VALVE	RX2
SUBT-	248	WESTINGHOUSE	AR440A	RELAY	TB2
SUBT-	261	AGASTAT	2412AE	RELAY	TB2
SUBT-	287	ROTORK	70A	VALVE ACTUATOR	RX2
SUBT-	305	GENERAL ELECTRI	HEA61A293	RELAY	TB2
SUBT-	320	ROTORK	30A	VALVE ACTUATOR	TB2
SUBT-	390	VICKERS	DG5S4-042AT-21	VALVE	TB2
SUBT-	391	VICKERS	DG5S4-042AE-21	VALVE	TB2
SUBT-	400	ASCO	8316B16	VALVE	RX2
SUBT-	466	ASCO	830081U	VALVE	TB2
SUBT-	487	ITE	EF3-B015	STARTER	TB2
SUBT-	495	MOOG	72-102	VALVE	TB2
SUBT-	509	ITE	HE3A-100	CIRCUIT BREAKER	TB2
SUBT-	511	VICKERS	DG5S4-C42AWB-40	VALVE	RX2
SUBT-	512	LIMITORQUE	SMB-4T	VALVE ACTUATOR	TB2
SUBT-	513	VICKERS	DG5S4-042AT-30	VALVE	RX2
SUBT-	514	VICKERS	DG5S4-042AT-31	VALVE	RX2
SUBT-	515	HI TEMP	71M1002	THERMAL HOOD	RX2
SUBT-	516	HI TEMP	71M1001	THERMAL HOOD	RX2

@@TERM

ATTACHMENT A

GENERAL ATOMIC COMPANY
REP80 BY SYSTEM

SAFETY RELATED TAGGED COMPONENTS
(RESPONSE TO BULLETIN IE 79-01B)

COMPONENT 1	SRB LOC 35	COMPONENT 1	SRB LOC 35	COMPONENT 1	SRB LOC 35
PSH- 11177	RX2	HSV- 2135-2	RX2	HV- 2190-4	RX2
PSH- 11178	RX2	LSV- 2136	RX2	HV- 2190-5	RX2
PSH- 11179	RX2	HSV- 2136-1	RX2	HV- 2190-6	RX2
PSH- 11180	RX2	LSV- 2136-1	RX2	HV- 2190-7	RX2
PSH- 11181	RX2	HSV- 2136-2	RX2	HV- 2190-8	RX2
PSH- 11182	RX2	LV- 2137	RX2	HV- 2190-9	RX2
PSH- 11183	RX2	LSL- 2137	RX2	HV- 2193	RX2
PSH- 11184	RX2	LV- 2138	RX2	HV- 2194	RX2
PSH- 11185	RX2	LSL- 2138	RX2	LSL- 21113	RX2
PSH- 11186	RX2	PDIS- 2155	RX2	PV- 21120	RX2
PSH- 11187	RX2	PDIS- 2156	RX2	LT- 21121	RX2
PSH- 11188	RX2	XEP- 2175	RX2	LT- 21122	RX2
P- 2101	RX2	XEP- 2176	RX2	LT- 21123	RX2
P- 2101-S	RX2	XEP- 2177	RX2	LT- 21124	RX2
P- 2102	RX2	XEP- 2178	RX2	LT- 21129	RX2
P- 2102-S	RX2	FIS- 2183	RX2	LS- 21130	RX2
P- 2103	RX2	FIS- 2184	RX2	LV- 21130	RX2
P- 2103-S	RX2	FIS- 2185	RX2	LSH- 21132	RX2
SV- 2105	RX2	FIS- 2186	RX2	FIS- 21137	RX2
ZS- 2105	RX2	HV- 2187-1	RX2	FIS- 21138	RX2
P- 2106	RX2	HV- 2187-2	RX2	FIS- 21139	RX2
SV- 2106	RX2	HV- 2187-3	RX2	FIS- 21140	RX2
ZS- 2106	RX2	HV- 2187-4	RX2	PDIS- 21149	RX2
P- 2107	RX2	HV- 2187-5	RX2	PDIS- 21150	RX2
P- 2108	RX2	HV- 2187-6	RX2	PDIS- 21151	RX2
SV- 2109	RX2	HV- 2187-7	RX2	PDIS- 21152	RX2
HV- 2109-1	RX2	HV- 2187-8	RX2	PDIS- 21153	RX2
HV- 2109-2	RX2	HV- 2187-9	RX2	PDIS- 21154	RX2
ZS- 2109-2	RX2	HV- 2188-1	RX2	PDIS- 21155	RX2
P- 2110	TB2	HV- 2188-2	RX2	PDIS- 21156	RX2
SV- 2110	RX2	HV- 2188-3	RX2	PDIS- 21157	RX2
HV- 2110-1	RX2	HV- 2188-4	RX2	PDIS- 21158	RX2
HV- 2110-2	RX2	HV- 2188-5	RX2	PDIS- 21159	RX2
ZS- 2110-2	RX2	HV- 2188-6	RX2	PDIS- 21160	RX2
SV- 2111	RX2	HV- 2188-7	RX2	PDIS- 21173	RX2
ZS- 2111	RX2	HV- 2188-8	RX2	PDIS- 21174	RX2
SV- 2112	RX2	HV- 2188-9	RX2	PDIS- 21175	RX2
ZS- 2112	RX2	HV- 2189-1	RX2	PDIS- 21176	RX2
SV- 2115	RX2	HV- 2189-2	RX2	PDIS- 21177	RX2
HV- 2115-1	RX2	HV- 2189-3	RX2	PDIS- 21178	RX2
HV- 2115-2	RX2	HV- 2189-4	RX2	PDIS- 21179	RX2
ZS- 2115-2	RX2	HV- 2189-5	RX2	PDIS- 21180	RX2
SV- 2116	RX2	HV- 2189-6	RX2	PDIS- 21181	RX2
HV- 2116-1	RX2	HV- 2189-7	RX2	PDIS- 21182	RX2
HV- 2116-2	RX2	HV- 2189-8	RX2	PDIS- 21183	RX2
ZS- 2116-2	RX2	HV- 2189-9	RX2	PDIS- 21184	RX2
LSV- 2135	RX2	HV- 2190-1	RX2	HV- 21185	RX2
HSV- 2135-1	RX2	HV- 2190-2	RX2	XEP- 21185	RX2
LSV- 2135-1	RX2	HV- 2190-3	RX2	HV- 21186	RX2

COMPONENT 1	SRB LOC 35	COMPONENT 1	SRB LOC 35	COMPONENT 1	SRB LOC 35
XEP- 21186	RX2	HV- 21416-1	RX2	PSL- 2231	TB2
HV- 21187	RX2	ZS- 21416-1	RX2	PSL- 2233	TB2
XEP- 21187	RX2	HV- 21416-2	RX2	PSL- 2235	TB2
HV- 21188	RX2	FT- 21425	RX2	HV- 2237	TB2
XEP- 21188	RX2	FT- 21426	RX2	HV- 2238	TB2
HV- 21213	RX2	FT- 21427	RX2	FT- 2239	RX2
HV- 21214	RX2	FT- 21428	RX2	FV- 2239	RX2
HV- 21257	RX2	HV- 2201	TB2	XEP- 2239	RX2
HV- 21258	RX2	ZS- 2201	TB2	FT- 2240	RX2
HV- 21259	RX2	HV- 2202	TB2	FV- 2240	RX2
HV- 21260	RX2	ZS- 2202	TB2	XEP- 2240	RX2
PT- 21285	RX2	HV- 2203	TB2	HV- 2241	RX2
PDT- 21285-1	RX2	ZS- 2203	TB2	HV- 2242	RX2
XEP- 21285-1	RX2	HV- 2204	TB2	PV- 2243	RX2
PDT- 21285-2	RX2	ZS- 2204	TB2	PV- 2244	RX2
XEP- 21285-2	RX2	FT- 2205	TB2	HV- 2249	RX2
PT- 21286	RX2	FV- 2205	TB2	ZS- 2249	RX2
PDT- 21286-1	RX2	FT- 2206	TB2	HV- 2250	RX2
XEP- 21286-1	RX2	FV- 2206	TB2	ZS- 2250	RX2
PDT- 21286-2	RX2	FT- 2209	TB2	HV- 2251	RX2
XEP- 21286-2	RX2	FT- 2210	TB2	ZS- 2251	RX2
FV- 21297	RX2	FT- 2211	TB2	HV- 2252	RX2
FSL- 21297	RX2	FT- 2212	TB2	ZS- 2252	RX2
FV- 21298	RX2	FT- 2213	TB2	HV- 2253	TB2
FSL- 21298	RX2	FT- 2214	TB2	HV- 2254	TB2
PDIS- 21319	RX2	HV- 2215	RX2	HV- 2265	TB2
PDIS- 21320	RX2	ZS- 2215	RX2	HV- 2266	TB2
PDIS- 21321	RX2	HV- 2216	RX2	PT- 2267	TB2
PDIS- 21322	RX2	ZS- 2216	RX2	XEP- 2267	TB2
PDIS- 21323	RX2	HV- 2217	RX2	PT- 2268	TB2
PDIS- 21324	RX2	ZS- 2217	RX2	PV- 2268	TB2
PDIS- 21325	RX2	HV- 2218	RX2	PSL- 2269	TB2
PDIS- 21326	RX2	ZS- 2218	RX2	PSL- 2271	TB2
PDIS- 21327	RX2	HV- 2223	TB2	PSL- 2273	TB2
PDIS- 21328	RX2	HV- 2224	TB2	HV- 2290	RX2
PDIS- 21329	RX2	TE- 2225-1	RX2	ZS- 2290	RX2
PDIS- 21330	RX2	TE- 2225-2	RX2	HV- 2291	RX2
PDIS- 21393	RX2	TE- 2225-3	RX2	ZS- 2291	RX2
PDIS- 21395	RX2	TE- 2225-4	RX2	HV- 2292	TB2
PDIS- 21396	RX2	TE- 2225-5	RX2	HV- 2293	TB2
PDIS- 21397	RX2	TE- 2225-6	RX2	ZS- 22115	RX2
PDIS- 21398	RX2	TE- 2226-1	RX2	ZS- 22116	RX2
PDT- 21411	RX2	TE- 2226-2	RX2	ZS- 22117	RX2
PDT- 21412	TB2	TE- 2226-3	RX2	ZS- 22118	RX2
PDT- 21413	RX2	TE- 2226-4	RX2	PV- 22129	TB2
PDT- 21414	TB2	TE- 2226-5	RX2	XEP- 22129	TB2
HV- 21415-1	RX2	TE- 2226-6	RX2	PT- 22129-1	TB2
ZS- 21415-1	RX2	PV- 2229	TB2	XEP- 22129-1	TB2
HV- 21415-2	RX2	PV- 2230	TB2	PV- 22130	TB2

ATTACHMENT A

ATTACHMENT A

COMPONENT 1	SRB LOC 35
XEP- 22130	TB2
PT- 22130-1	TB2
XEP- 22130-1	TB2
HV- 22131	TB2
HV- 22132	TB2
HV- 22133	TB2
HV- 22134	TB2
TE- 22135	RX2
TE- 22136	RX2
TE- 22137	RX2
TE- 22138	RX2
TE- 22139	RX2
TE- 22140	RX2
TE- 22142	RX2
TE- 22143	RX2
TE- 22144	RX2
TE- 22146	RX2
PV- 22153	TB2
XEP- 22153	TB2
XEP- 22153-1	TB2
PV- 22154	TB2
XEP- 22154	TB2
XEP- 22154-1	TB2
PSH- 22197	TB2
PSH- 22198	TB2
HV- 22200	RX2
HV- 22201	RX2
HV- 22202	RX2
HV- 22203	RX2
HV- 22204	RX2
HV- 22205	RX2
HV- 22206	RX2
HV- 22207	RX2
HV- 22208	RX2
HV- 22209	RX2
HV- 22210	RX2
HV- 22211	RX2
HV- 22212	RX2
HV- 22213	RX2
HV- 22221	TB2
HV- 22222	TB2
HV- 22223	TB2
HV- 22224	TB2
HV- 22225	TB2
HV- 22226	TB2
HV- 22227	TB2
HV- 22228	TB2
HV- 2366-1	RX2
HV- 2366-2	RX2

COMPONENT 1	SRB LOC 35
PSL- 3123	TB2
PSL- 3124	TB2
PSL- 3125	TB2
HV- 4225	TB2
HV- 4257	TB2
PS- 4266	TB2
HSV- 4266	TB2
TE- 4637-3	RX2
TE- 4638-3	RX2
L- 8201	TB2
S- 8201	TB2
C- 8201-S	TB2
S- 8202	TB2
L- 8203	TB2
PS- 8207	TB2
PS- 8208	TB2
TS- 8208	TB2
PS- 8214	TB2
TS- 8214	TB2
TS- 8218	TB2
TS- 8219	TB2
PS- 8221	TB2
PS- 8222	TB2
TS- 8236	TB2
TS- 8237	TB2
PS- 8244	TB2
TS- 8245	TB2
TS- 8246	TB2
PS- 8247	TB2
PS- 8248	TB2
TS- 8249	TB2
PS- 8258-1	TB2
PS- 8258-2	TB2
HS- 9101-2	RX2
P- 9101-SX	RX2
P- 9101-X	RX2
HS- 9102-2	RX2
P- 9102-SX	RX2
P- 9102-X	RX2
HS- 9103-2	RX2
HS- 9104-2	RX2
P- 9105-X	RX2
P- 9106-X	RX2
LSL- 91147	RX2
LSL- 91148	RX2
N- 9208	TB2

COMPONENT 1	SRB LOC 35
N- 9219	TB2
N- 9220	TB2
N- 9231	RX2
I- 9321-A	RX2
I- 9321-B	RX2
I- 9321-C	RX2
I- 9321-D	RX2
I- 9354	TB2
I- 9355	TB2
I- 9371	TB2
I- 93123	RX2
I- 93124	RX2
I- 93125	RX2
I- 93126	RX2
I- 93127	RX2
I- 93128	RX2
I- 93129	RX2
PDSH- 93129	RX2
I- 93130	RX2
PDSH- 93130	RX2
I- 93131	RX2
PDSH- 93131	RX2
I- 93132	RX2
PDSH- 93132	RX2
I- 93133	RX2
PDSH- 93133	RX2
I- 93134	RX2
PDSH- 93134	RX2
I- 93135	RX2
I- 93136	RX2
I- 93137	RX2
I- 93138	RX2
I- 93139	RX2
I- 93140	RX2
I- 93141	RX2
I- 93142	RX2
I- 93143	RX2
TSH- 93448	RX2
TSH- 93449	RX2
TSH- 93450	RX2
TSH- 93451	RX2
TSH- 93452	RX2
TSH- 93453	RX2
XE- 93454-A	RX2
XE- 93454-B	RX2
XE- 93454-C	RX2
XE- 93455-A	RX2
XE- 93455-B	RX2
XE- 93455-C	RX2

COMPONENT 1	SRB LOC 35	COMPONENT 1	SRB LOC 35	COMPONENT 1	SRB LOC 35
XE- 93456-A	RX2				
XE- 93456-B	RX2				
XE- 93456-C	RX2				
XE- 93457-A	RX2				
XE- 93457-B	RX2				
XE- 93457-C	RX2				
XE- 93470-A	RX2				
XE- 93470-B	RX2				
XE- 93470-C	RX2				
XE- 93471-A	RX2				
XE- 93471-B	RX2				
XE- 93471-C	RX2				
TE- 93472	RX2				
TE- 93473	RX2				
TE- 93474	RX2				
XE- 93479-A	RX2				
XE- 93479-B	RX2				
XE- 93479-C	RX2				
XE- 93480-A	RX2				
XE- 93480-B	RX2				
XE- 93480-C	RX2				

ATTACHMENT A

ALPHA PREFIXES

Designation	Description
A	Absorbers, Traps and Demineralizers
C	Compressors, Blowers, Vacuum Pumps, Fans Including Drives
E	Exchangers, Cooling Towers
F	Filters, Strainers, and Dryers
I	Instrument and/or Control Racks and Panels
N	Electrical Power/Control Cabinets
P	Pumps and Drives
S	Special Packaged Items
T	Tanks and Vessels
FV	Flow Valve
FT	Flow Transmitter
HS	Hand Switch
HV	Hand Valve
LS	Level Switch
LT	Level Transmitter
LV	Level Valve
PS	Pressure Switch
PT	Pressure Transmitter
PV	Pressure Valve
SV	Speed Valve
TS	Temperature Switch
XE	Special Element (Steamline Rupture Sensor)
ZS	Position Switch
FIS	Flow Indicating Switch
FSL	Flow Switch Low
HS	Hand Solenoid Valve
LSH	Level Switch High
LSL	Level Switch Low
LSV	Level Solenoid Valve
PDT	Pressure Differential Transmitter
PSH	Pressure Switch High
PSL	Pressure Switch Low
TSH	Temperature Switch High
XEP	Special Electrical Pneumatic Transducer
PDIS	Pressure Differential Indicating Switch
PDSH	Pressure Differential Switch High

SYSTEM NUMBERS

System	Description
11	Reactor Vessel and Internal Components
21	Primary Coolant System
22	Secondary Coolant System
23	Helium Purification System
31	Feedwater and Condensate
42	Service Water System
46	Reactor Plant Cooling Water System
82	Instrument and Service Air
91	Piping-Hydraulic Oil System
92	Electrical-Including Switchgear and Standby Diesel Generator
93	Control and Instrumentation

Section 2.1.6.b -- Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used In Post-Accident Operations

PSC December 12, 1979 (P-79299) REPLY:

"PSC will perform the radiation protection design reviews required by Section 2.1.6.b, utilizing the source terms recommended in Regulatory Guides 1.3, 1.4, and 1.7, and will submit the results of the review to the NRC by January 1, 1980. Where doses received are in excess of GDC 19 guidelines, PSC will take those steps necessary to permit post-accident operations in vital areas. Any required modifications will be completed by January 1, 1981."

PSC December 27, 1979 (P-79312) SUBMITTAL:

The assessment of post-accident operator actions in vital areas at Fort St. Vrain (FSV) indicates that doses received from a hypothetical FSV accident scenario will not be in excess of the GDC 19 guidelines for the duration of the accident, provided the FSV reactor plant exhaust filters are adequately shielded.

PSC hereby commits to providing necessary shielding modifications to the FSV reactor plant exhaust filters by January 1, 1981 to permit operator access to vital areas under accident conditions.

The hypothetical Fort St. Vrain (FSV) accident scenario consists of the FSV Design Basis Accident (DBA) #1 combined with successive PCRV primary coolant leakage after depressurization. For clarification, the DBA #1 and PCRV leakage scenarios are explained below.

DISCUSSION:

To obtain a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3, 1.4 and 1.7 requires a permanent loss of all forced circulation for the FSV HTGR. This specific accident was identified as DBA #1 in FSAR Section 14.10 and Appendix D. These analyses performed by General Atomic Company at the time of licensing did not consider Regulatory Guides 1.3 and 1.4 source terms (i.e., the equivalent of the 50% of the core radioiodine and 100% of the core noble gas inventory for release to the primary coolant) appropriate for the HTGR. However, because of past precedence by the then Atomic Energy Commission (AEC) of using the above source terms, offsite doses resulting from the postulated accident were calculated and presented in the previously mentioned FSAR sections using both the General Atomic Company release assumptions and AEC TID-14844 release assumptions. In both cases the offsite doses are within 10CFR100 limits.

DBA #1 Description:

A non-mechanistic loss of forced circulation is postulated from full power operation, where the reactor is scrammed by the plant protection system and all attempts to restore forced circulation using the multiple heat sinks, circulators and motive power for the circulators fail. Because of the large heat sink provided by the graphite core, considerable time is available to initiate primary coolant depressurization and to restore forced circulation. The FSV FSAR specifies the time available to initiate depressurization to be 5 hours, which was later amended by PSC letter P-77250 dated December 22, 1977 to be 2 hours. The reduction in time was due to the capability of the helium purification system to process primary coolant during the planned blowdown of the clean primary coolant to the reactor building ventilation stack. Thus, the depressurization of the PCRV is initiated after 2 hours and completed 7 hours later (or 9 hours from the onset of the accident), at which time the PCRV has been depressurized to 5 psig.

The fuel is slow to heat up due to the large heat sink provided by the core graphite. A peak average active core temperature of 5400°F is reached about 80 hours after the onset of the accident. At this temperature, the core structural integrity and geometry are not compromised since the vaporization temperature of graphite is 6900°F. Peak activity released to the primary coolant, considering decay, is reached about 24 hours into the accident.

Heat removal is provided by the liner cooling system in the redistribute mode which maximizes cooling in the top head of the PCRV.

Leakage of primary coolant from the PCRV is assumed to occur at a conservatively high leakage rate of 0.2% of the primary coolant inventory per day.

Offsite doses were calculated for a 6 month duration of the accident, but most of the offsite dose occurs in the first 200 hours of the accident, due to fission product decay.

The reactor building ventilation system maintains continuous venting of the reactor building environment at 1.5 volumes/hr during the entire period of the accident.

Primary Coolant Leakage Rate During DBA #1:

The FSV FSAR DBA #1 (Appendix D, page D.1-56) assumed an arbitrarily conservative and non-mechanistic estimate of PCRV leakage after the intentional depressurization by assuming that the liner has failed completely (or does not exist) and only concrete permeability controls the leakage. An internal 5 psi pressure differential was assumed which purportedly gave a PCRV leak rate of 8.33×10^{-5} fraction per hr (0.2%/day). Reference was made to Question IX.7 of Amendment No. 9 of the FSV FSAR for the calcualtion of the permeation rate for the FSV PCRV concrete under these conditions.

Examination of Question D.2 revealed simply the conclusion that a 5 psi positive differential pressure led to 0.2%/day and 2 psi positive differential pressure led to 0.08%/day. Question IX.7 also did not provide details of the calculation of the 0.2%/day rate. However, considerable detail and a derivation was provided for the analysis of leakage rate tests at high pressures. The following equation was provided (eqn.14 on page IX.7-8):

$$W\ (\text{lb/day}) = 1.13 \times 10^{-5} \frac{\Delta P}{\Delta P_c} \frac{A}{X} \ln\left(\frac{P_1}{P_2}\right) + 2.2 \times 10^{-6} \frac{\Delta P}{\Delta P_c} \frac{A}{X} \left(P_1^2 - P_2^2\right) \quad \text{(eqn. 14)}$$

Where ΔP = PCRV inside pressure in psig

ΔP_c = PCRV inside pressure in psig for which the net compressive stress in concrete = 0

A = Face area of concrete, ft^2

X = Concrete thickness, ft

P_1 = Permeation or high side pressure, psia

P_2 = Ambient or low side pressure, psia

Numerical values were inserted for P_1 = 845 psig with the assumption that ΔP_c was approximately equal to P_1 in the following equation (eqn.15 on same page):

$$W = 1.13 \times 10^{-5} \times \frac{9000}{10} \ln \frac{857.5}{12.5} + 9.1 \times 10^{-7} \frac{9000}{10} \left(857.5^2 - 12.5^2\right)$$

$$= 0.043 + 602 = 600 \text{ lb/day} \quad \text{(eqn. 15)}$$

The first item to note is that the coefficient for the second (laminar flow) term is in error which is most likely a single error in transcribing from equation 14 to 15 since equation 13 has the 9.1×10^{-7} coefficient. Equation 15 should read:

$$W = 1.13 \times 10^{-5} \times \frac{9000}{10} \ln \frac{857.5}{12.5} + 2.2 \times 10^{-6} \frac{9000}{10} \left(857.5^2 - 12.5^2\right)$$

$$= 0.043 + 1445 = 1450 \text{ lb/day} \quad \text{(eqn. 15 revised)}$$

The second item is that the $\Delta P/\Delta P_c$ term has been dropped in going from eqn.14 to eqn.15, which is significant if it is assumed that these equations are appropriate for evaluating the leak rate at P_1 = 5 psig.

LEAK RATE

Pressure P_1 (psig)	lb/day Eqn 14	15	15 Revised	%/day 14	15	15 revised	Given
							App D; Amend 9 Question D.2
5	.0019	.13	.30	.001	.07	.17	.20
							Amend 9 Question D.2
2	.0003	.046	.107	.0001	.025	.059	.08

Since equation 14 is the appropriate equation, the 0.2%/day leak rate is conservative by a factor of 200. Furthermore, the only equation that comes close to the values given in the SAR is 15 Revised, that is, $\Delta P/\Delta P_o$ has been neglected which accounts for the factor of 200.

For purposes of plant shielding and equipment environmental evaluations, the historic 0.2%/day is assumed to exist as an upper limit of all potential contaminated primary coolant leakage including permeability through the PCRV concrete. This is judged to be conservative since the primary coolant with any significant activity is contained within the PCRV or helium purification components contained in wells within the PCRV.

Radionuclide Source Terms for DBA-1:

As previously stated, the fuel within the graphite core is slow to heatup during DBA#1. Once it has reached the FSAR fuel particle coating failure temperature of 1725°C (3137°F), the fission products are assumed, for purposes of this shielding evaluation, to be realeased per the TID-14844 assumptions. For release to the primary coolant within the PCRV, this is 100% of noble gases, 50% of the iodines and 1% others. The total activity in curies contained in the primary coolant, assuming no leakage from the PCRV, as a function of lapsed time, is given in Table 2.1.6.b-1.

Consistent with TID-14844 release assumptions, 50% of the iodines plateout within the primary coolant system resulting in a depletion of the iodine to 25% of core inventory in the reactor building air. Thus, the total activity in curies in the reactor building, assuming the upper limit of 0.2%/day leakage (which is being purged by the reactor building ventilation system at the rate 1.5 volumes/hr), is given in Table 2.1.6.b-2.

TABLE 2.1.6.L-1

-NUREG-0578 STUDY TOTAL ACTIVITY (Ci) PRESENT IN THE PCRV PRIMARY COOLANT AT GIVEN ELAPSED TIME (hours). PCRV PRESS BOUNDARY REMAINS INTACT. TID-14844 NORMALIZATION FRACTIONS USED, 100% NOBLE GASES, 50% IODINE 1% OTHERS

ELAPSED TIME (Hours)

UCLIDE	2	8	24	34	40	48	52	58	72	120	240	475	720	4320
-88	1.05+04	2.89+05	2.80+05	2.39+04	5.89+03	1.37+03	5.50+02	1.76+02	7.04+01	0	0	0	0	0
-88	8.57+03	2.79+05	2.80+05	2.66+04	6.51+03	1.46+03	6.07+02	1.89+02	7.08+01	0	0	0	0	0
-95	3.15+01	6.66+03	1.84+05	2.57+05	3.01+05	3.59+05	3.69+05	3.84+05	4.18+05	4.12+05	3.88+05	3.43+05	3.02+05	4.60+0
-95	3.18+01	6.74+03	1.87+05	2.63+05	3.09+05	3.69+05	3.80+05	3.97+05	4.35+05	4.37+05	4.31+05	4.12+05	3.88+05	8.90+0
-131	1.33+03	3.50+05	6.18+06	6.91+06	7.33+06	7.88+06	7.90+06	7.93+06	7.98+06	7.57+06	4.89+06	2.07+06	8.45+05	0
-132	1.44+03	2.34+05	1.79+06	6.09+05	5.64+05	5.61+05	3.68+05	2.96+05	2.72+05	1.76+05	4.02+04	4.99+03	5.46+02	0
-133	2.48+03	5.30+05	6.44+06	5.25+06	4.70+06	4.12+06	3.65+06	3.05+06	2.04+06	4.84+05	8.81+03	0	0	0
-133	5.25+03	1.40+06	2.50+07	2.78+07	2.94+07	3.14+07	3.14+07	3.12+07	3.09+07	2.73+07	1.41+07	3.86+06	9.90+05	0
-135	1.98+03	2.46+05	1.40+06	5.49+05	3.31+05	1.88+05	1.25+05	6.83+04	1.78+04	2.94+02	0	0	0	0
-135M	7.28+02	8.34+04	4.59+05	1.72+05	1.04+05	5.97+04	3.91+04	2.14+04	5.58+03	0	0	0	0	0
-135	1.75+03	5.43+05	6.24+06	3.86+06	2.93+06	2.11+06	1.62+06	1.08+06	4.39+05	1.81+04	0	0	0	0
-140	5.44+01	1.44+04	2.58+05	2.92+05	3.13+05	3.39+05	3.42+05	3.45+05	3.54+05	3.57+05	2.70+05	1.56+05	8.80+04	0
-140	3.34+01	7.37+03	2.01+05	2.60+05	2.93+05	3.36+05	3.43+05	3.54+05	3.75+05	3.96+05	3.10+05	1.80+05	1.01+05	0

TABLE 2.1.6.b-2

NUREG-0578 STUDY TOTAL ACTIVITY (Ci) PRESENT IN THE REACTOR BUILDING ATMOSPHERE AT GIVEN ELAPSED TIME (hours). PCRV LEAK RATE TO BUILDING 0.2%/DAY. REACTOR BUILDING VENTED AT 1.5 VOLUMES/HR. TID-14844 NORMALIZED FRACTIONS USED, 100% NOBLE GASES, 25% IODINE, 1% OTHERS

CLIDE	ELAPSED TIME (Hours) 2	8	24	34	40	48	52	58	72	120	240	475	720	4320
-88	3.77-01	1.31+01	1.33+01	1.32+00	3.22-01	7.10-02	3.00-02	9.23-03	3.38-03	0	0	0	0	0
-88	3.55-01	1.37+01	1.42+01	1.48+00	3.58-01	7.77-02	3.34-02	1.02-0.	3.61-03	0	0	0	0	0
-95	1.20-03	3.29-01	9.81+00	1.40+01	1.64+01	1.97+01	2.04+01	2.12+01	2.31+01	2.29+01	2.16+01	1.91+01	1.68+01	2.56+00
-95	1.21-03	3.33-01	9.98+00	1.43+01	1.69+01	2.02+01	2.10+01	2.19+01	2.41+01	2.43+01	2.39+01	2.29+01	2.16+01	4.94+00
131	2.52-02	8.64+00	1.65+02	1.90+02	2.02+02	2.17+02	2.19+02	2.20+02	2.21+02	2.10+02	1.36+02	5.76+01	2.35+01	0
132	2.57-02	5.24+00	4.24+01	1.58+01	1.46+01	1.46+01	1.05+01	8.46+00	7.75+00	5.14+00	1.30+00	1.61-01	1.77-02	0
133	4.68-02	1.30+01	1.70+02	1.44+02	1.29+02	1.13+02	1.01+02	8.45+01	5.65+01	1.34+01	2.45-01	0	0	0
-133	1.99-01	6.94+01	1.34+03	1.54+03	1.63+03	1.75+03	1.75+03	1.75+03	1.75+03	1.52+03	7.85+02	2.14+02	5.50+01	0
35	3.68-02	5.89+00	3.60+01	1.51+01	9.05+00	5.09+00	3.47+00	1.89+00	4.89-01	0	0	0	0	0
135M	3.14-02	7.71+00	8.59+01	7.38+01	5.53+01	3.53+01	2.75+01	1.81+01	6.20+00	1.07-01	0	0	0	0
135	6.75-02	2.73+01	3.40+02	2.26+02	1.72+02	1.22+02	9.56+01	6.40+01	2.56+01	1.01+00	0	0	0	0
140	2.06-03	7.12-01	1.38+01	1.61+01	1.72+01	1.87+01	1.89+01	1.91+01	1.96+01	1.98+01	1.50+01	8.67+00	4.89+00	0 -
140	1.27-03	3.66-01	1.08+01	1.43+01	1.61+01	1.85+01	1.90+01	1.96+01	2.08+01	2.20+01	1.72+01	9.98+00	5.62+00	0

Radiation Levels During DBA-1:

Based upon TID-14844 source term release assumptions, the radiation levels were calculated in the reactor building and the control room to determine the operator accessibility. Details are described herein.

Assumptions

In addition to the assumptions used in deriving the source terms, the following assumptions were made for evaluating shielding adequacy:

1. Credit was taken for a 50% depletion of the iodines due to plateout in the primary coolant system prior to release to the reactor building atmosphere.

2. All fission products were assumed to remain gasborne. In other words, no plateout of fission products was contemplated.

3. All the activities were uniformly distributed throughout the free space of the reactor building or the PCRV.

4. The iodines and particulates removed by the reactor-building ventilation filters were deposited in any two of the three filters available.

5. Only major shielding such as concrete walls was considered.

Reactor Building

To determine the accessibility of the reactor building during the course of DBA-1, the gamma dose rate in the reactor building was calculated as a function of elapsed time. The contributing sources consist of the gasborne activity in the reactor building as a result of PCRV leakage, the primary coolant activity contained in the PCRV, and the buildup of iodines and particulates on the reactor building ventilation HEPA and charcoal adsorbers. The contribution from the ventilation filters was not considered, as the filters will be properly shielded.

Two dose points were selected for the dose-rate calculation. The first point is located at the center of the space above the refueling floor (≈ 40 ft from the floor), and the second point is on the refueling floor directly above the refueling penetration. The PATH code described in FSAR Section 11.2.2.4 was utilized to perform the calculation.

Figure 1 shows the dose rate at the first dose point. Essentially all the contributions come from the gasborne activity in the reactor building. The activity in the PCRV is relatively insignificant to the first dose point, because of a large separation distance between the source and dose point. Short-term access to the reactor building is possible.

The dose rate at the second dose point (i.e., on the refueling floor) is given in Figure 2. The contributions from the reactor building and from the PCRV are individually represented, along with the total dose rate. The contribution from the PCRV is due to the primary coolant activity present in the interspace below the primary closure for the control rod drive. The maximum dose rate on the floor is 1.0 rem/hr, which is less than the peak dose rate of 1.4 rem/hr at the first dose point. Therefore, the refueling floor is accessible on a short-term basis.

Control Room

The dose rates in the control room include the contributions from the airborne activity in the reactor building atmosphere, and from the iodine and particulate activity accumulated in the plant ventilation filters. The PATH code was used to determine the contribution from each source as a function of time into accident. The dose point was located in the reactor engineer's office, as shown in Figure 3.

The results of the PATH calculations are shown in Figure 3 as a function of elapsed time. It is apparent that the contribution from the airborne activity in the reactor building is relatively small or negligible as compared with that from the reactor building ventilation filters. The dose rate reaches a peak of 700 mrem/hr about one month into accident. The important nuclides are Zr95, Nb95 and La140 accumulated in the filters.

The dose rate in the control room appears to be excessive for continuous manned access. Adequate shielding will be provided for the ventilation filters so that the dose rate from the filters can be reduced to an acceptable level.

Summary Results

The peak dose rates in the reactor building and control room are summarized below. Also indicated are the time at which the peak dose rate occurs following an accident, and the total dose accumulated over a period of 180 days from the initiation of the accident.

Location & Condition	Peak Gamma Dose Rate	Time of Peak	180 Day Accumulated Dose (rem)
Reactor Building (above refueling floor)	1.4 R/hr	24 hrs.	400
Control Room			
From Vent Filters (Unshielded)	700 mR/hr	≃ 720 hrs.	2400
From Reactor Building	3 mR/hr	24 hrs.	0.9

Conclusion

The following conclusions are reached from the review of shielding design adequacy for DBA-1 conditions and TID-14844 source term release assumptions:

1. The reactor building ventilation filters will be adequately shielded to reduce the dosage contribution from the filters.

2. Areas immediately outside the reactor building should be accessible only on a restricted basis, because of direct radiation from the activity in the reactor building.

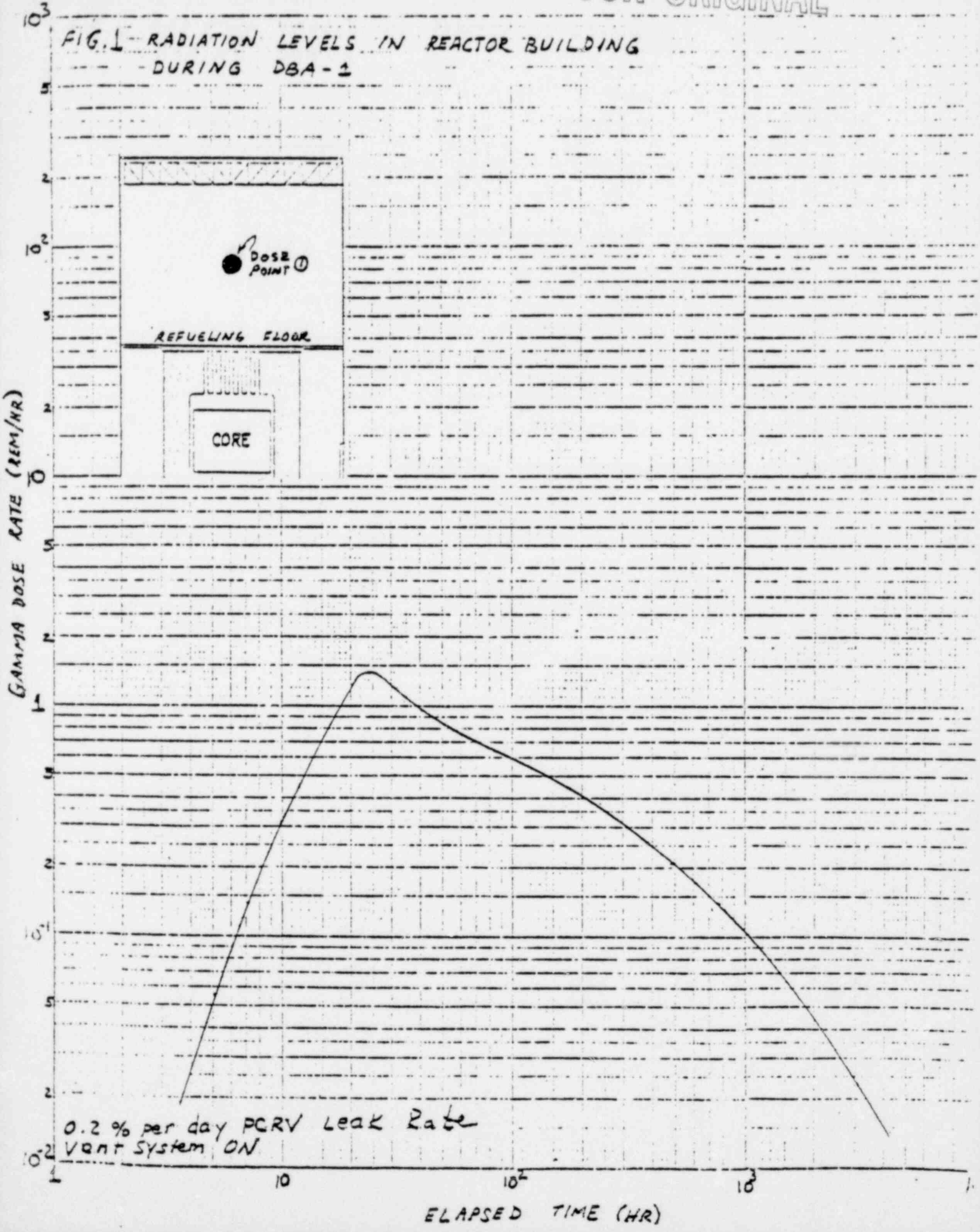

FIG. 1 RADIATION LEVELS IN REACTOR BUILDING DURING DBA-1

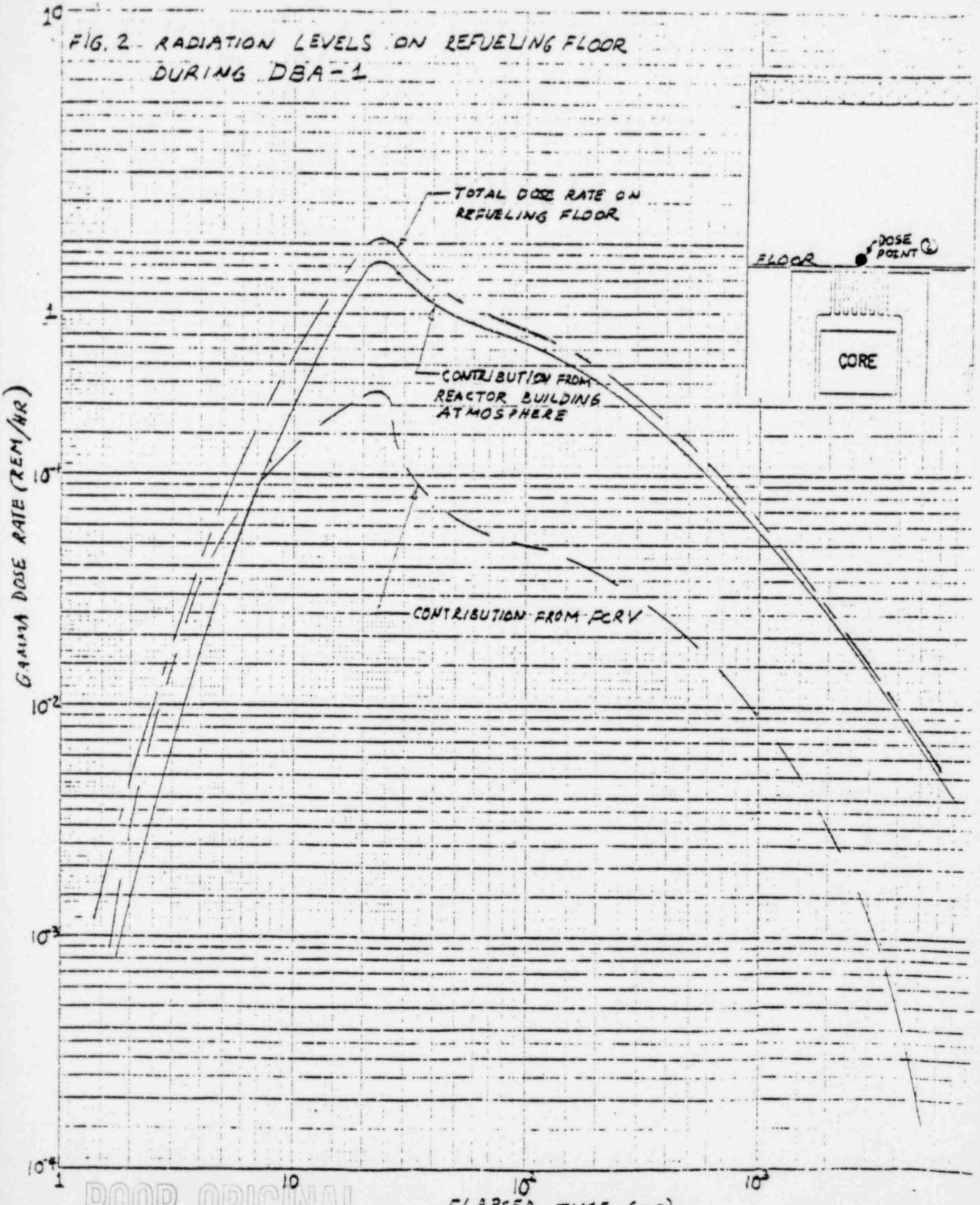

FIG. 2 RADIATION LEVELS ON REFUELING FLOOR DURING DBA-1

FIG. 3 RADIATION LEVELS IN CONTROL ROOM DURING DBA-1