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Advanced Technology Application Station Blackout Core Damage Frequency Reduction – The Contribution of AN AC Independent Core Residual Heat Removal System

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Abstract. An event of station blackout (SBO) can result in severe core damage and undesirable consequences to the public and the environment. To cope with an SBO, nuclear reactors are provided with protection systems that automatically shut down the reactor, and with safety systems to remove the core residual heat. In order to reduce core damage frequency, the design of new reactors incorporates passive systems that rely only on natural forces to operate. This paper presents an evaluation of the SBO core damage frequency of a PWR reactor being designed in Brazil. The reactor has two core residual heat removal systems - an AC dependent system, and a passive system. Probabilistic safety assessment is applied to identify failure scenarios leading to SBO core damage. The SBO is treated as an initiating event, and fault trees are developed to model those systems required to operate in SBO conditions. Event trees are developed to assist in the evaluation of the possible combinations of success or failure of the systems required to cope with an SBO. The evaluation is performed using SAPHIRE, as the software for reliability and risk assessment. It is shown that a substantial reduction in the core damage frequency can be achieved by implementing the passive system proposed for the LABGENE reactor design.
Keywords: Station blackout, passive safety system, core damage frequency.

1. INTRODUCTION

One of the main areas in the analysis of severe accidents in nuclear reactors is that of station blackout. The loss of offsite electrical power concurrent with turbine trip and unavailability of the onsite emergency ac power system can result in loss of decay heat removal capability, leading to potential core damage.

Attempts to estimate SBO frequencies were first reported in WASH-1400 [1]. It was shown, for example, that the risk of core-melt associated to SBO was an important contributor to the total risk of the Surry Station, and should not be ignored as a potential initiating event of severe accidents.

The results reported in WASH-1400 along with questions related to the reliability of emergency diesel generators, and the number of nuclear plants that experienced a total loss of off-site power, led the NRC to raise the SBO as an unresolved safety issue in 1979. This status ended in 1988 with the publication of a new rule, as stated in paragraph 50.63 - 10 CFR50 [2], and in the Regulatory Guide 1.155 “Station Blackout” [3]. In order to comply with the SBO rule, many existing nuclear power plants have improved their safety systems. New emergency electric sources were added, the capacity of the batteries was increased and mechanically driven pumps were incorporated into safety systems.

In August 2003, fifteen years after SBO Rule being in place, the United States Nuclear Regulatory Commission published a report on the “Regulatory Effectiveness of the Station Blackout Risk at Nuclear Power Plants” [4]. The report shows the results-for the United States Nuclear Power Plants- of the measures carried out as required by the “SBO Rule” (10 CFR 50.63 – 1988). The “Rule is effective and the industry and the Nuclear Regulatory Commission costs to implement it were reasonable considering the outcome”, was reported in NUREG-1776. The report also shows that in twenty, out of forty-six PWR plants, the station blackout contributes with 20 %, or more to the core damage frequency.

More recently, in 2005, the NRC published NUREG/CR-6890[5], an update version to previous reports. The report examines events of loss of offsite power and the associated station blackout core damage risk at the US commercial nuclear power plants. The results presented in NUREG/CR-1776 and in NUREG/CR-6890 are summarized in Table I. Although improvements in safety are observed as a result of the SBO rule, it can be seen that SBO events can still be a significant contributor to core damage frequency in existing nuclear power plants.

The present paper examines the SBO CDF for the LABGENE reactor. The contribution of a passive residual heat removal system in reducing the SBO CDF is investigated. The study was limited to the identification and quantification of accident sequences that potentially could lead to SBO core damage at the LABGENE reactor. Systems and devices involved in an SBO condition are described and their influence on the accident sequences is discussed. Event trees and fault trees are developed, and each accident sequence is discussed in some detail. Two design alternatives are examined. Quantification is performed using the SAPHIRE reliability and risk assessment software [6]. The results are compared with those presented in NUREG/CR-6890.

2. THE LABGENE NUCLEAR REACTOR

The LABGENE is a 48 MW thermal PWR prototype reactor being designed in Brazil. The reactor is intended to serve as a test bed for developing the capability to design small and medium power reactors for electricity production, and for nuclear propulsion. The total cost of LABGENE is estimated to be around US$ 488 million. So far US$ 318 million have been spent in this project. Some of the main components of the reactor have been delivered to the site, namely the pressure vessel and its internals, the pressurizer, the two steam generators, the primary coolant pumps, and components of the secondary circuit [7]. The reactor site is located in a rural area about 120 km from the city of São Paulo, Brazil.

The reactor has two primary coolant loops. The pressure vessel, steam generators, the primary pumps, and the pressurizer are enclosed in a steel containment, which is surrounded by a water tank used as a shielding pool and also as a heat sink. A confinement building houses the steel containment and a secondary system having two turbo-generators. Facilities for spent fuel storage, waste treatment, and auxiliary systems are located in adjacent buildings.

Table 1. Number of United States Nuclear Power Plants in SBO CDF Range

	Nuclear Power Plants in SBO CDF Range											
SBO CDF range (x 10^{-5} per reactor-year) – see note 2	< 0.5	0.5 0.99	1.0 1.49	1.5 1.99	2.0 2.49	2.5 2.99	3.0 3.49	3.5 3.99	4.0 4.49	4.5 4.99	5.0 9.99	10 35
Before SBO rule implementation (estimated)	5	13	14	7	13	4	9	5	4	3	13	10

Expected after SBO rule implementation	23	23	14	9	6	5	6	5	4	0	5	0
Actual outcome after SBO rule implementation (as in NUREG-1776)	46	22	13	17	1	3	1	3	0	1	1	0
Actual outcome after SBO rule implementation (as in NUREG/CR-6890)	85	10	3	5	0	0	0	0	0	0	0	0

NOTES:

1- The expected numbers were defined in NUREG-1109 (1988).

2- The SBO CDF in NUREG/CR-6890 is per reactor critical year

The LABGENE reactor has two independent systems to remove decay heat from the core: A forced circulation AC dependent system (FCR), and a natural circulation system (NCR) that exchange decay heat with the shielding pool. As shown later in this paper, the NCR system plays an important role in reducing the SBO core damage frequency.

Some features of the systems required to operate in an SBO condition at the LABGENE reactor are described below:

Electric Power System

A commercial nuclear power plant operates interconnected with an external grid in order to supply electricity. This external grid will provide the required power to the plant auxiliary systems following a reactor shutdown. To face random failures, which could simultaneously disable both internal and external electric sources, commercial nuclear power plants are also provided with a redundant external electric source.

The LABGENE is not intended to supply energy to the electrical grid. The reactor is interconnected to only one external source, which is equivalent to the second source of commercial nuclear power plants. This external source is responsible for supplying AC electric power during startup and after the reactor shutdown.

In order to meet regulatory requirements of high reliability electric supply, LABGENE electric power circuits (FIG. 1) has two local emergency diesel electric sources for each of the two existing load trains. Each single emergency source is designed to have sufficient capacity to feed the total expected demand.

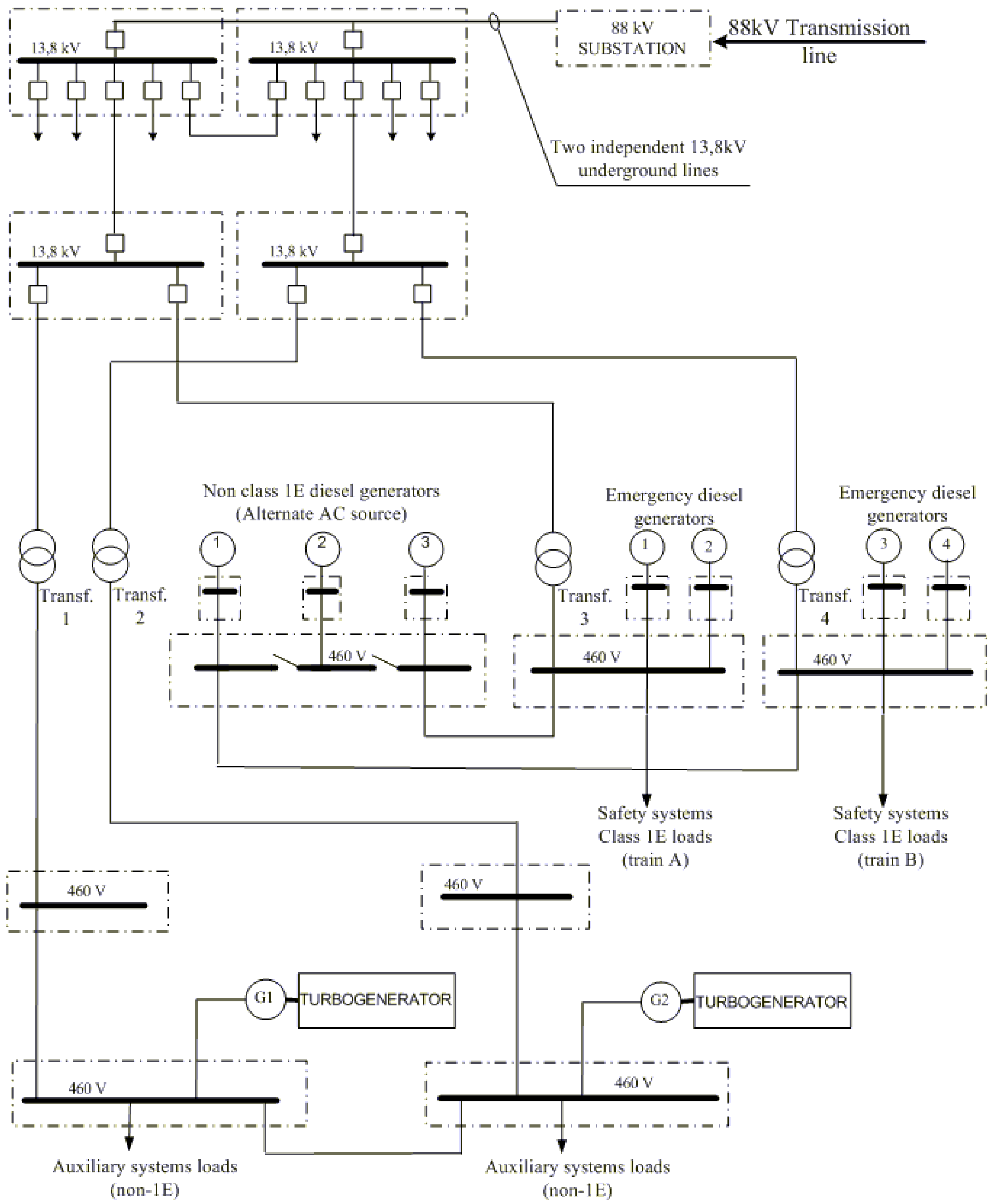


Fig. 1. LABGENE electric power simplified one line diagram

The idea of increasing the number of local emergency electric sources to compensate for a relatively greater vulnerability of external electric supplies is not new. Borst [8], in his comments related to the GDC 17 (USNRC, 10CFR50 - 1988), wrote that in Germany "*it has been realized that it is either impossible or not economically feasible at most locations, where large nuclear power plants would be permitted in the country, to provide two adequate power line connections from truly independent electrical systems*", and that it is "*more important to concentrate on securing the greatest possible reliability of the onsite standby and emergency power systems, rather than to depend to any large extent on the dubious reliability of a second source of offsite power*" .

Alternate AC Power Source

An Alternate AC power source (AAC) – as defined in § 50.2 of 10 CFR50 [2] – is an onsite (or nearby) electric power source, available to cope with a station blackout, provided it meets the following regulatory requirements:

(1) Is connectable to but not normally connected to the offsite or onsite emergency AC power systems;
(2) Has minimum potential for common mode failure with offsite power or the onsite emergency AC power sources;
(3) Is available in a timely manner after the onset of station blackout; and
(4) Has sufficient capacity and reliability for operation of systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).

Alternate AC sources, as defined in 10 CFR50 § 50.2, "*will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate ac power source(s) and this equipment shall be demonstrated by test"……" If the alternate ac source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required*" (10 CFR 50 /02/).

LABGENE has three diesel generators capable to function as AAC power sources. At normal operating conditions these three AC sources will remain on standby. They will automatically start on demand to assume loads of the essential auxiliary systems (non-class 1E). For an SBO condition, however, one of the three generators will be commanded to leave its normal function and will be switched to supply the electric power required by safe shutdown loads and core residual heat removal systems.

DC Electric Power System

The Direct Current (DC) electric power systems are very important to nuclear plants. They are designed to supply energy to loads important to recover the plant from an SBO condition. The LABGENE reactor has two batteries of DC power supply. Each battery is designed with sufficient capacity to supply energy for a period of time greater than the expected duration of an SBO. This period of time can also be considerably extended if restrictions to the use of DC power are adequately imposed.

Emergency Heat Sink

The shielding pool of the LABGENE reactor can be used as an emergency heat sink. In an SBO condition the pool is capable to function as a heat sink to remove the residual core heat for a period of time much greater than the time expected for the SBO duration (about forty times).

Core Residual Heat Removal Systems

The LABGENE reactor does not have an AC independent Feed Water System to supply water to the steam generators. Nevertheless, it has two Core Residual Heat Removal Systems: A Forced Core Residual Heat Removal System (FCR), which is AC dependent; and a Natural Convection System (NCR), which is a passive cooling system.

The FCR system has two independent and redundant trains. Both trains have pumps driven by AC induction motors. The main components and interconnections of this system are schematically represented in FIG. 2.

The NCR system (FIG. 3) has a heat exchanger which transfers heat from the primary circuit to the shielding pool. The system is AC independent, but it has four solenoid valves driven by the DC Electric System. The electric power to operate these valves is supplied by the emergency batteries, assumed to be operational during an SBO condition. These valves are of the bistable type, which means that they only need electric energy for a short period of time during their opening pulse. After opening, the valves will stay open without need of any further electrical supply. These valves can have their operation commanded either automatically or manually.

3. STATION BLACKOUT AT LABGENE

Station blackout at LABGENE reactor is defined by the following sequence of events: loss of electric power provided by the turbo-generators, concurrent with the unavailability of on-site power (four class 1E diesel generators), and the loss of the off-site power sources.

Important features of LABGENE associated with a station blackout condition are summarized below:

- There is only one off-site power supply (external grid);

- The turbo-generators do not feed the external grid, so they are not affected by external grid transients;

- There are two emergency diesel generators for each load train; each DG has enough capacity to power the total emergency loads;

- No AC independent pumps for feeding water to the steam generators;

- No Containment Cooling System (that in general is AC dependent);
- There is an AC independent Core Residual Heat Removal System (NCR); and
- The primary cooling water pumps are of canned type. Leaks to the containment are not expected to occur (Leaks resulting from the loss of pump seal cooling are known to be a factor that increases the probability of core uncovering at commercial nuclear power plants in case of an SBO [9].

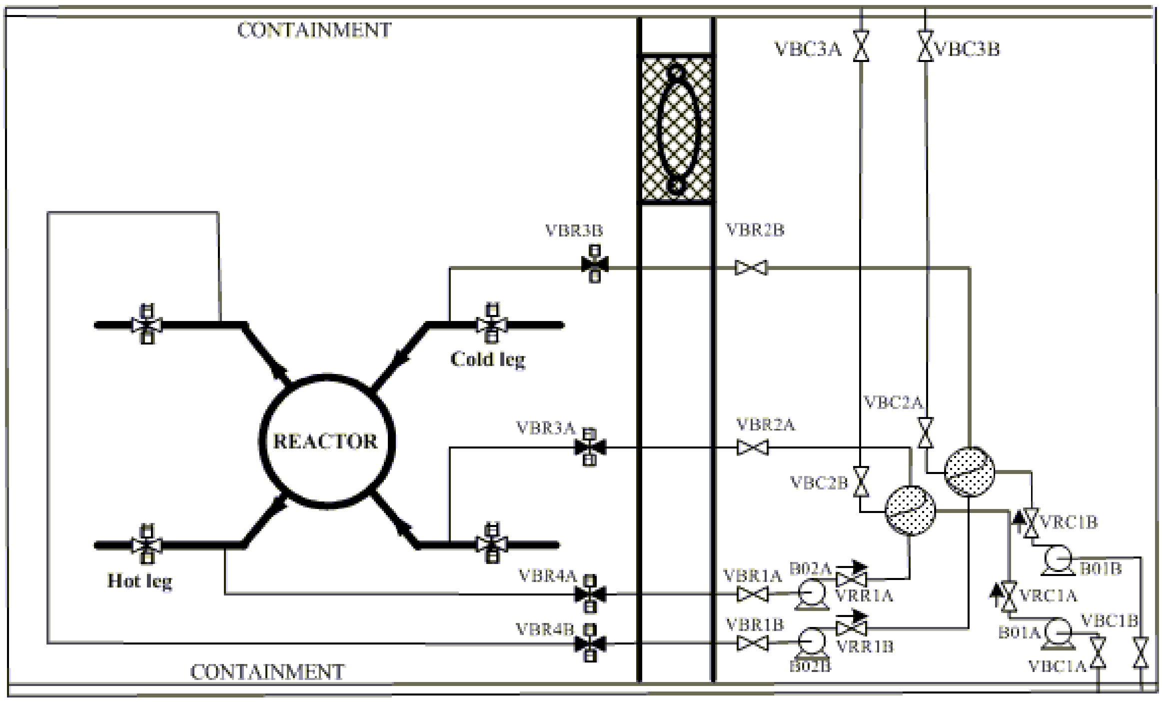


Fig. 2. Forced Circulation Core Residual Heat Removal System(FCR)

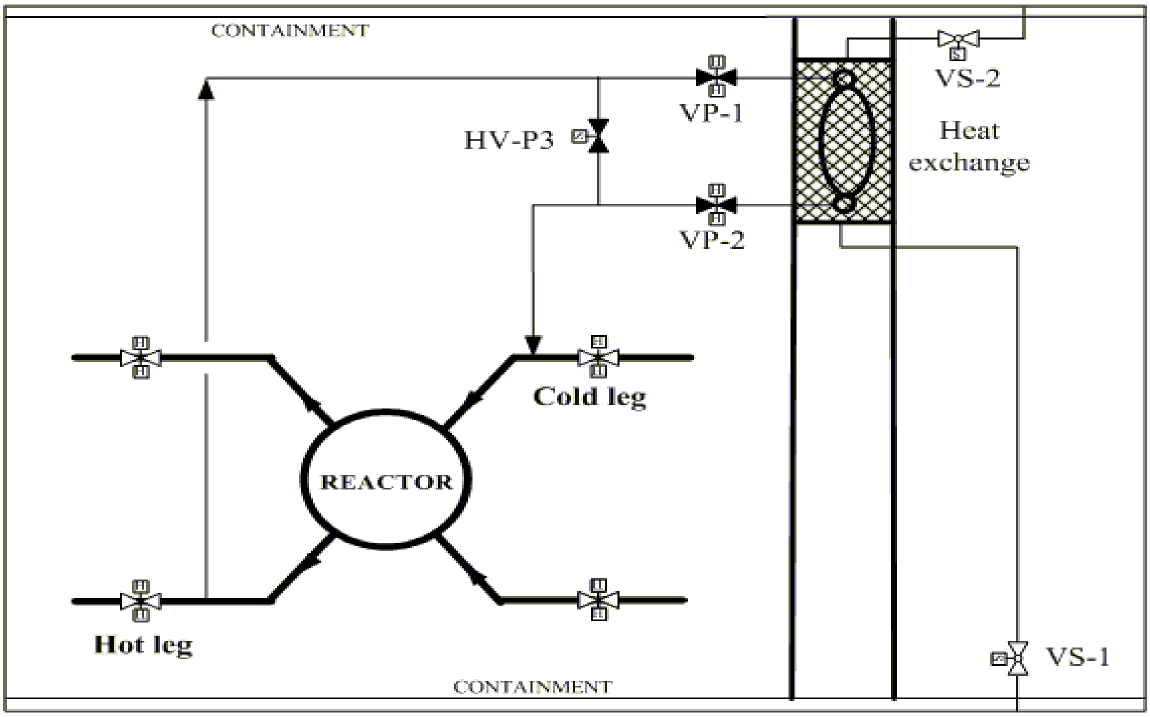


Fig. 3. Natural Circulation Core Residual Heat Removal System (NCR)

4. QUANTIFICATION OF SBO CDF

The model developed to quantify the SBO CDF assumes the LABGENE reactor operating at full power. It is based on data available on the Preliminary Safety Analysis Report and data from thermal hydraulic simulations for accident conditions. A team of experts in thermal hydraulic and electricity helped to identify components and systems responsible for preventing or mitigating accidents originated by an SBO. Having identified the SBO accident sequences, they were quantified using fault trees and event trees. The SAPHIRE computer code was used to quantify the sequences.

It was postulated that an SBO would not happen simultaneously with design basis accidents, as they are considered to be independent events [10]. The probability of a LOCA or an ATWS occurring simultaneously with an SBO is therefore ignored.

The SBO frequency was estimated using the fault tree shown in FIG. 4, which includes the trip of the two turbo-generators, concurrent with the unavailability of both onsite emergency ac power provided by the diesel generators and the loss of the off-site power supply.

The unavailability of the off-site power supply was determined based on eleven years of historical data for the 88 kV transmission line connected to the substation that supplies the LABGENE reactor site.

The trip of the two turbo-generators was included as a basic event in the fault tree. A very conservative frequency of twenty trips per year was assumed in the evaluation.

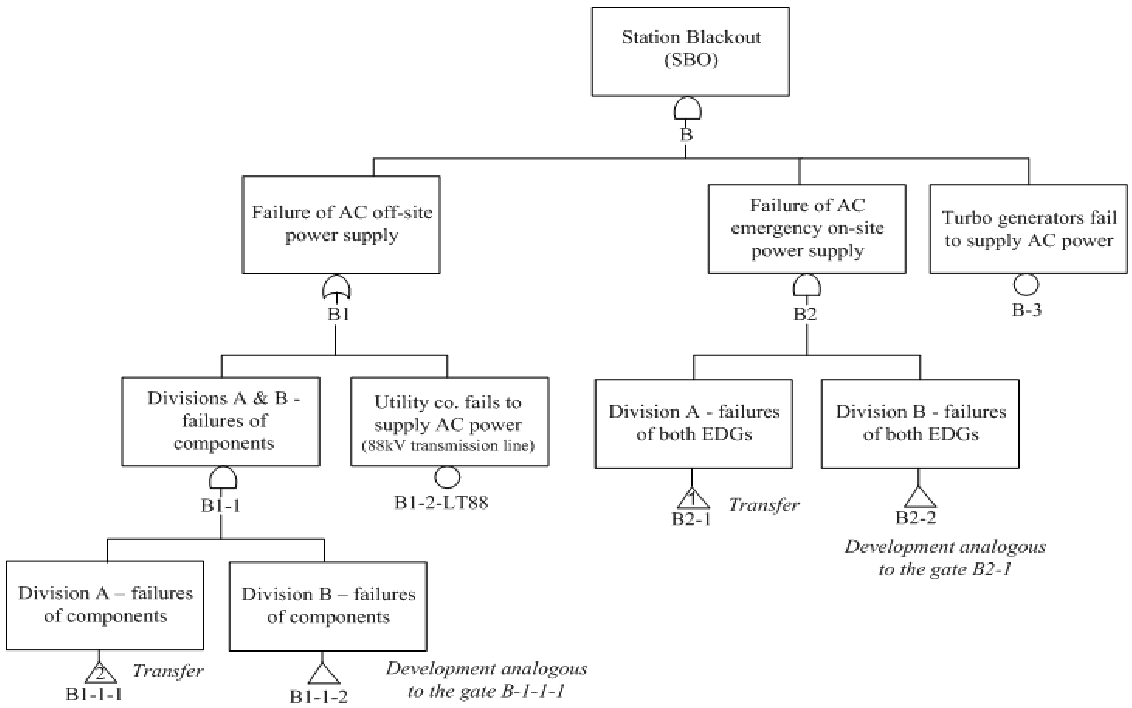
The contribution of the diesel generators failure to the SBO CDF was estimated by developing a fault tree based on the one line diagram shown in FIG. 1. Common cause failures were assumed possible to occur between diesel generators in the same train. Common cause events were modeled using data available in the PSA of Angra I Nuclear Power Plant, and the method described by Misra [11].

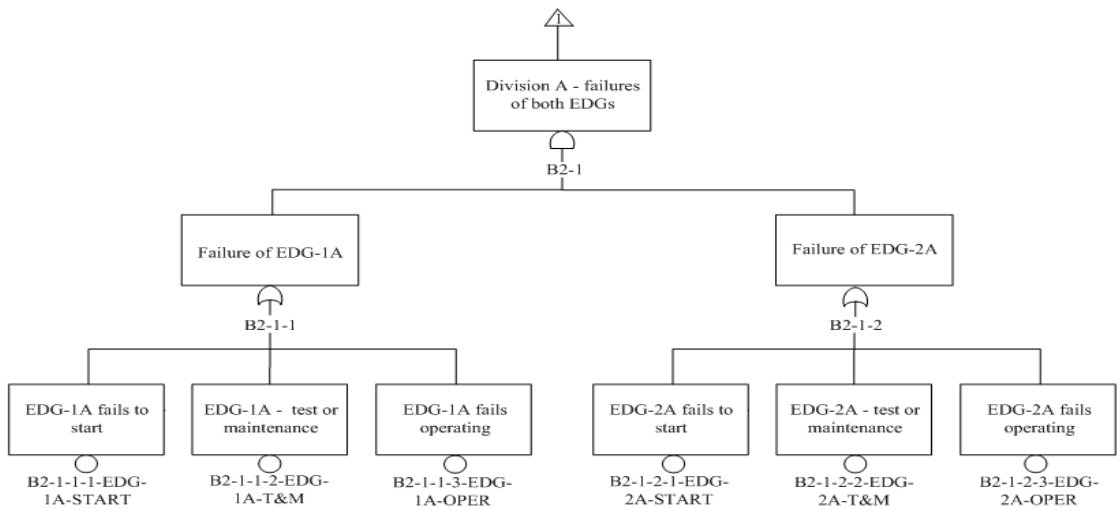
Except for the external electric supply, for which specific failure data was available, basic events data are generic. Data were taken from EGG-SSRE-8875 [12] and from the PSA of Angra 1 Nuclear Power Plant [13].

Recovery actions by the operators to restore failed components or systems were added directly into the event trees. The non-recovery probabilities were based on “*Probabilistic Safety Analysis Procedure Guide*“, NUREG/CR-2815 [14]. Table II summarizes operator actions and the associated probabilities used.

Two event trees were examined. FIG. 5 shows the event tree associated with the NCR and FCR system. In FIG 6 only the FCR system is included.

Fault trees were developed for each event shown in FIG. 5 and 6. This study shows only a simplified fault trees for to the natural convection system NCR (FIG. 7)





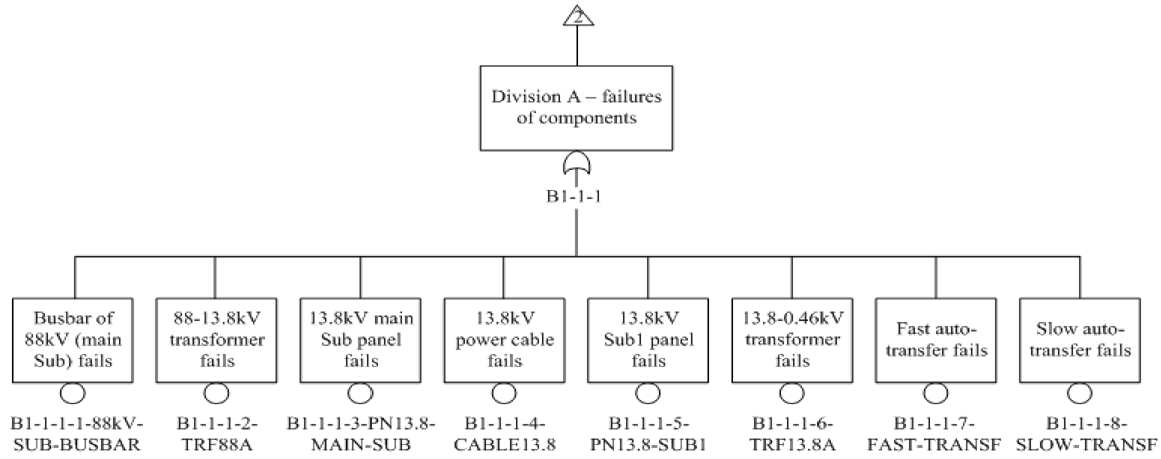
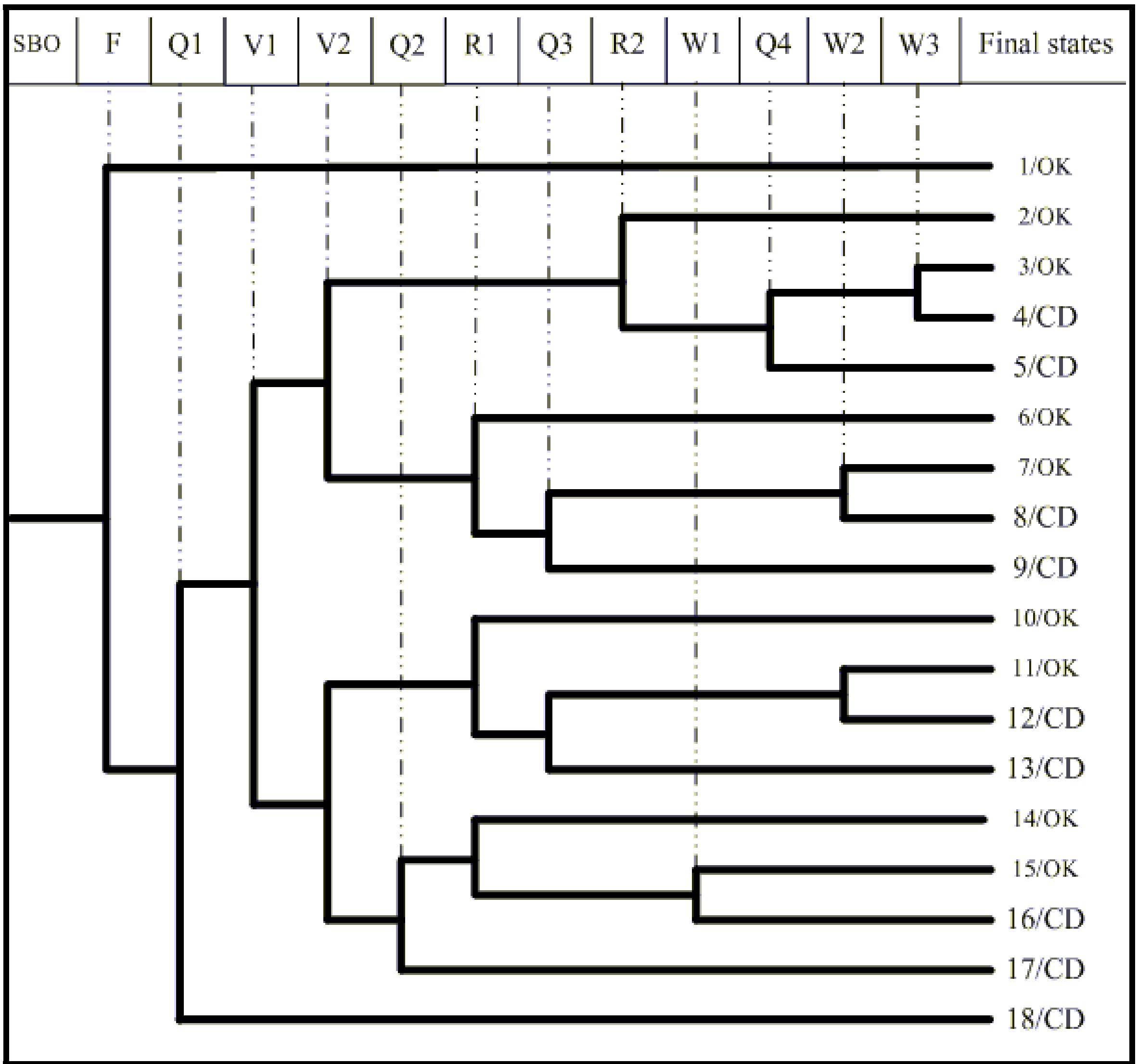


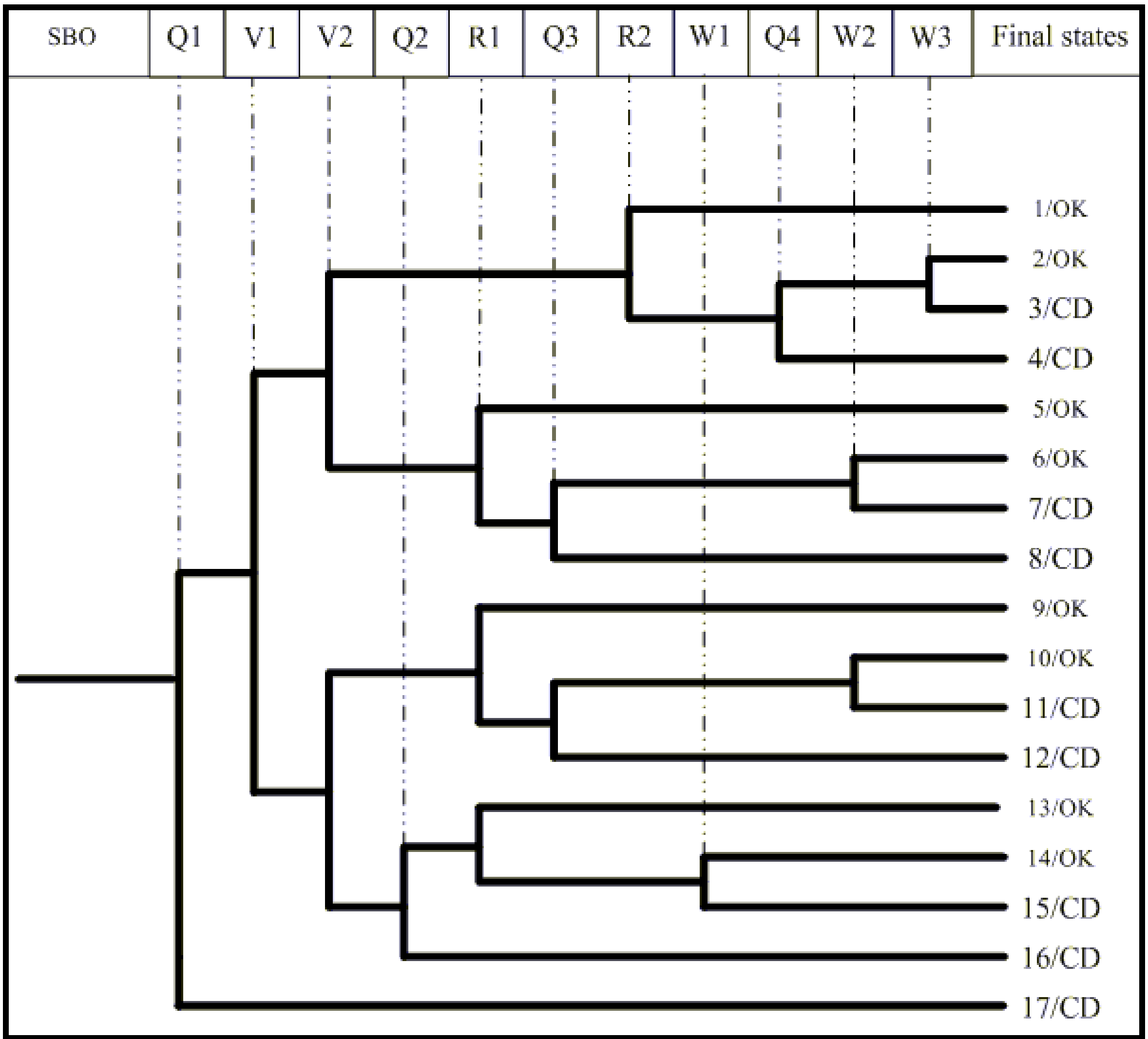
Fig. 4. Station Blackout Fault Tree



Event Acronym	Event description
SBO	Station blackout
F	Core residual heat removal by the NCR system
Q1	Initial pressure relief by pressurizer

V1	Steam Generator number one pressure relief
V2	Steam Generator two pressure relief
Q2	Pressurizer provides additional pressure relief
R1	Lining up FCR system 40 minutes after the SBO
Q3	Pressure relief by the pressurizer after 40 minutes following the SBO
R2	Lining up FCR system 80 minutes after the SBO
W1	AC recovering after 80 minutes of SBO
Q4	Pressure relief by the pressurizer after 80 minutes following the SBO
W2	AC recovering after 120 minutes of SBO
W3	AC recovering after 120 minutes of SBO

Fig. 5. Event tree with NCR and FCR core residual heat removal systems



Event Acronym	Event description
SBO	Station blackout
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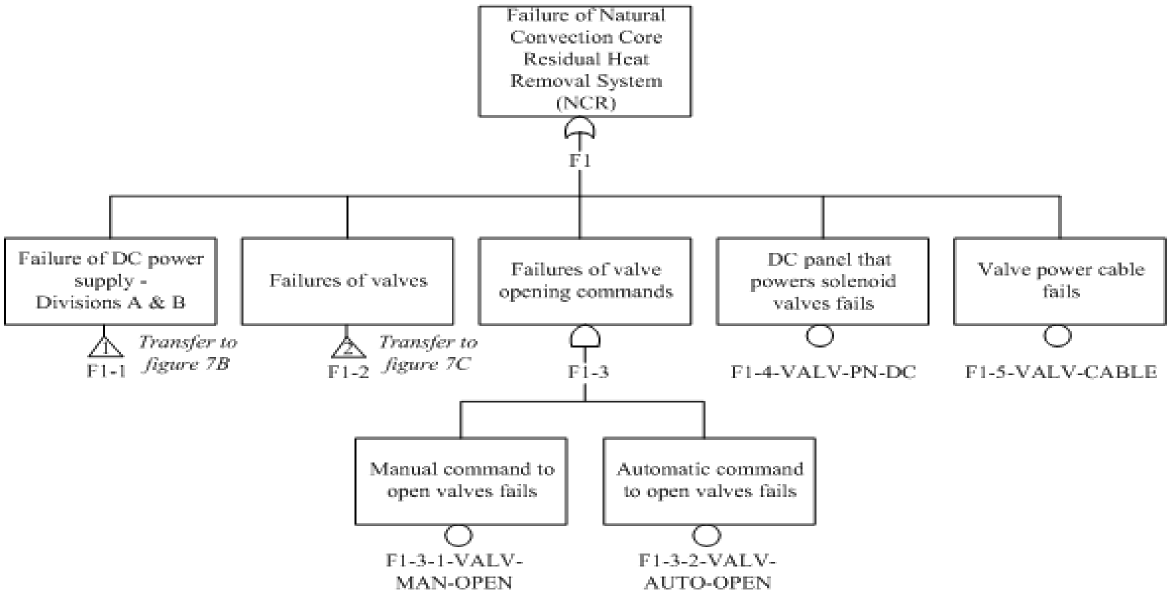
V2	Steam Generator two pressure relief
Q2	Pressurizer provides additional pressure relief
R1	Lining up FCR system 40 minutes after the SBO

Q3	Pressure relief by the pressurizer after 40 minutes following the SBO
R2	Lining up FCR system 80 minutes after the SBO
W1	AC recovering after 80 minutes of SBO
Q4	Pressure relief by the pressurizer after 80 minutes following the SBO
W2	AC recovering after 120 minutes of SBO
W3	AC recovering after 120 minutes of SBO

Fig. 6. Event tree without NCR core residual heat removal system



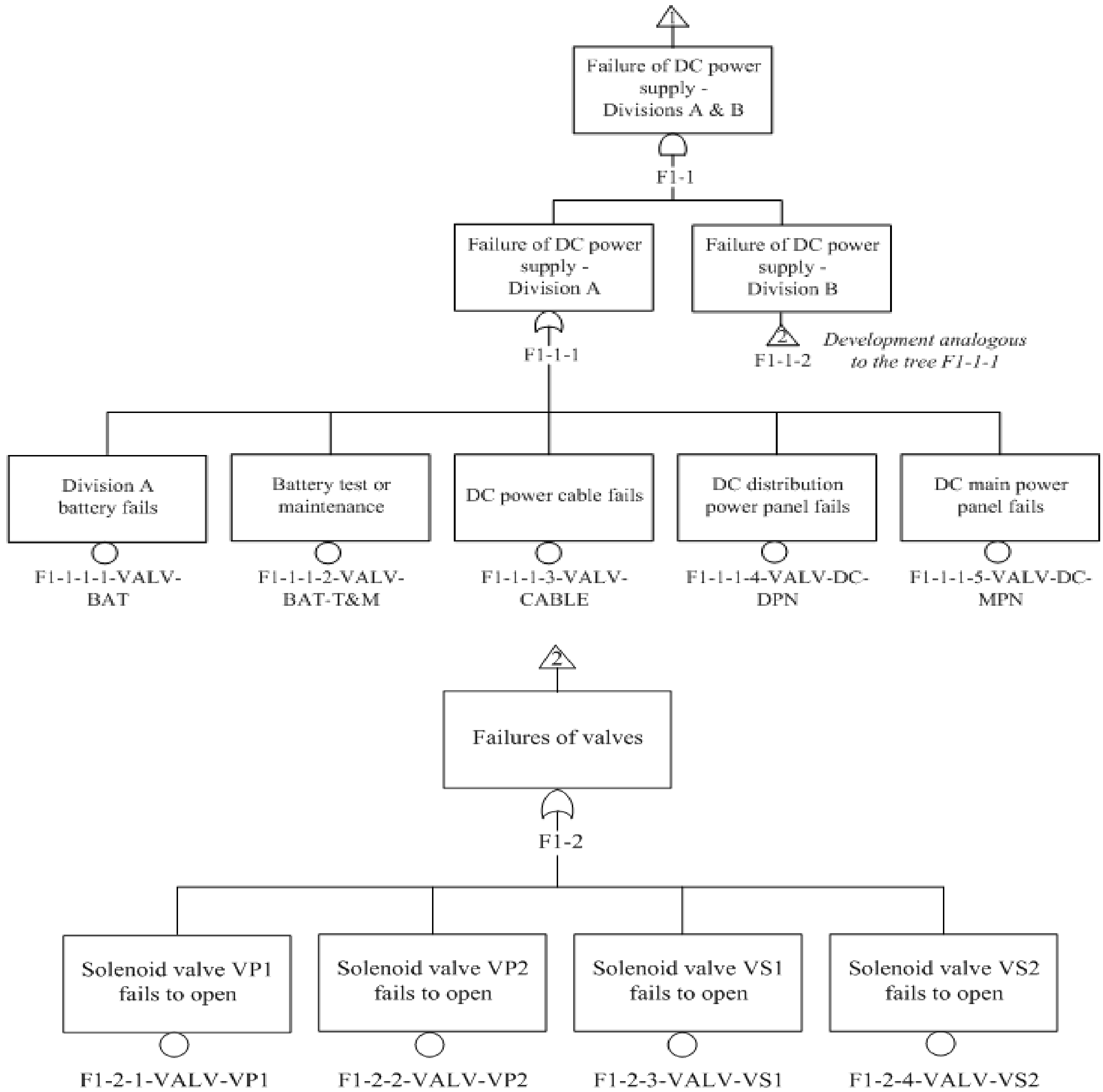


Fig. 7. Fault Tree of Natural Convection Core Residual Heat Removal System

Table 2. Summary of operator failure probabilities assumed for LABGENE SBO condition

Human failure description	Probability*	Comments
1. Failure to manually open the valves of the natural circulation core residual heat removal system (NCR).	0.1	As the failure would occur in the beginning of the reactor shutdown, it is assumed that the operators are under medium stress.
2. Failure to close a pressurizer relief valve stuck open	0.5	If the action is not successful the core will be uncovered; so the operators will be under great stress.

3. Failure to align the AAC in less than 20 minutes.	0.5	It is necessary to "island" a diesel generator and insert a circuit breaker; these actions demand skilled operators able to work under great stress.
4. Failure to align the AAC in less than 60 minutes.	0.1	The action is similar to the previous item 3. However, in this case there is more time to execute it.
5. Failure to start the forced circulation core residual heat removal system (FCR).	0.05	The action is not complex and will not be executed under great stress.

* Based on NUREG/CR-2815

Events Trees

The sequences 4, 5, 8, 9, 12, 13, 16, 17 and 18 listed in FIG. 5, describe the conditions, system successes, and system failures that would lead to Core Damage (CD). These sequences are described as follows:

Sequence 4 – The AAC power source does not operate and AC is not recovered (R2 event); the pressurizer relief valve operates successfully (Q4 event); AC from external source, or from emergency diesel generators, is not recovered before 160 minutes from the start of the SBO condition (W3 event). The FCR system cannot be lined up and the sequence final state is CD.

Sequence 5 – The AAC power source does not operate and AC is not recovered (R2 event); the pressurizer relief valve sticks open (Q4 event) and the core is uncovered.

Sequences 8 e 12 – Both sequences are similar and share the same frequency. Sequence eight is started by the relief valve of the Steam Generator Two sticking open (V2 event). Sequence twelve is started by a similar failure in the relief valve of the Steam Generator One (V1 event); the AC power from AAC power source is not recovered (R1 event); the pressurizer relief valve operates successfully (Q3 event) giving enough time to recover AC from external source, or from emergency diesel generators. However, in this case, the AC power is recovered only after 120 minutes from the start of the SBO condition (R1 and W2 events). The FCR system cannot be lined up and sequence final state is CD.

Sequences 9 e 13 - Both sequences are similar and their frequencies are equal. Accident sequence nine is started by the unsuccessful event V2 and sequence thirteen by the unsuccessful event V1; AC from the AAC power source is not recovered (R1 event); the pressurizer relief valve sticks open (Q3 event). The sequences final state is core uncovered.

Sequence 16 – The relief valve of the pressurizer produces a successful initial pressure relief (Q1 event). However, the relief valves of both steam generators fail to operate (V1 and V2 events); the pressurizer relief valve keeps alleviating the primary circuit pressure (Q2 event). Success of the Q2 event gives time to recover AC power, but the AC power is recovered only after 80 minutes from the start of the SBO condition (R1 and W1 events). The FCR system cannot be lined up and the sequence final state is core damage.

Sequence 17 – The relief valve of the pressurizer produces a successful initial pressure relief (Q1 event). However the relief valves of both steam generators fail to operate (V1 and V2 events); the pressurizer relief valve sticks open (Q2 event). The sequence final state is core uncovered.

Sequence 18 – Following an SBO, the pressurizer relief valve sticks open (Q1 event) and the sequence final state is core uncovered.

5. RESULTS AND CONCLUSIONS

The initiating event SBO, as modeled by the fault tree shown in FIG 4, has an estimated frequency of 7.1E-05 per year. A major contributor to this frequency is the loss of turbo generators, an event which is followed by the shutdown of the reactor. As LABGENE is a prototype, a conservative number of shutdowns were assumed in the analysis.

The electric power supply from the external grid is the second major contributor. However, it has been observed that in recent years the rate of supply interruptions has decreased substantially. If this tendency is maintained, a revision of the present analysis should show a correspondent decrease in the SBO frequency.

The calculated SBO frequency and the probabilities of the top events are presented in Table III. As seen from this Table the failure of the NCR system has a relatively low probability to occur. FIG. 5 shows that no core damage will occur if the NCR system succeeds in removing decay heat (F1 event).

Table IV and V show the calculated frequencies for scenarios that potentially could result in core damage. The results show that the proposed natural convection system for the LABGENE reactor can reduce the core damage frequency by a factor of about 500 times.

The SBO CDF uncertainty associated with the basic data was examined. Basic event data are generally inaccurate either because they are rare or because specific data is simply not available. The failure probabilities of the basic data, grouped according to their particular correlation, were assumed to be random variables distributed as lognormal curves. A lognormal curve was generated for the SBO CDF using the SAPHIRE computer program. FIG. 8 shows results in terms of mean, median and percentiles. The results clearly indicate the influence of the NCR system in reducing the SBO CDF for the LABGENE reactor.

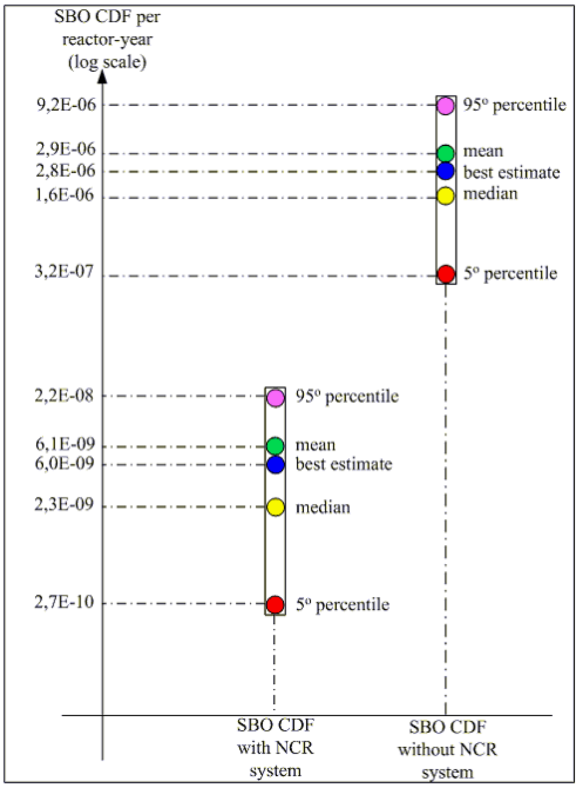


Fig. 8. SBO-CDF LABGENE

A sensitivity analysis was performed to identify systems, components, or actions that most influence the SBO CDF. If EDG performance to run is improved by a factor of ten - EDG failure probability to run divided by ten - the SBO CDF decreases from 6.0E-09 to 6.6E-10. If EDG performance to run is degraded by a factor of ten, the SBO CDF increases from 6.0E-09 to 1.8E-06. This behavior can be explained by examining the cut sets of the SBO fault tree (FIG.4). The occurrence of an SBO requires common cause events and failure combinations of the four EDG. The SBO CDF can be increased by powers of two, tree or four, depending on the number of combinations. Reducing the EDG reliability does not reduce the SBO CDF to the same extent as other type of failures will become more significant contributors. A similar evaluation was made for EDGs performance to start. The results, however, does not identify a sensitivity as high as in the previous case.

Table 3. Frequency of the initiating event and probabilities of the other top events in the event trees

Event acronym	Failure event description	Frequency or probability
SBO	Station blackout	7.1E-5 per reactor-year
F	NCR system fails to remove core residual heat	2.1E-3
Q1	Pressurizer valve sticks open after its first relief following SBO	1.5E-3
V1	Relief valve of number one steam generator sticks open before FCR system can be lined up	1.1E-1
V2	Relief valve of number two steam generator sticks open before FCR system can be lined up	1.1E-1
Q2	Pressurizer valve starts relieving just after SBO but sticks open before FCR can be lined up	9.4E-2
R1	The FCR system is not be lined up after 40 minutes following SBO (either AC from any source is not recovered or the FCR system is failed)	5.1E-1
Q3	Pressurizer valve starts relieving 40 minutes after SBO but sticks open before FCR can be lined up	9.3E-2
R2	FCR system is not be lined up after 80 minutes following SBO (either AC from any source is not recovered or FCR system is failed)	1.1E-1
W1	AC is not recovered (neither from off-site supply nor from emergency DGs) after 80 minutes following SBO	1.1E-1
Q4	Pressurizer valve starts relieving 80 minutes after SBO but sticks open before FCR can be lined up	9.2E-2
W2	AC is not recovered (neither from off-site supply nor from emergency DGs) after 120 minutes following SBO	7.9E-2
W3	AC is not recovered after 160 minutes following SBO	4.6E-2

Table 4. Frequencies of each accident sequence and total SBO CDF - Event tree without NCR system

Accident sequence number	Frequency (events per reactor-year)
3	3.8E-7
4	7.5E-7
7	3.2E-7
8	3.8E-7
11	3.2E-7
12	3.8E-7
15	4.9E-8
16	8.4E-8
17	1.1E-7
TOTAL	2.8E-6

Table 5. Frequencies of each accident sequence and total SBO CDF - Event tree with NCR system

Accident sequence number	Frequency (events per reactor-year)
4	8.1E-10
5	1.6E-9
8	7.0E-10
9	8.2E-10
12	7.0E-10
13	8.2E-10
16	1.0E-10
17	1.8E-10
18	2.3E-10
TOTAL	6.0E-9

Changes in the SBO CDF due to changes in the EDG outages for tests and maintenance were examined. The results show that increasing the number of outages - by reducing the reliability by a factor of ten- the SBO CDF increases proportionally. On the other hand, decreasing the number of outages- by increasing the reliability by a factor of ten- the SBO CDF decreases only four times.

Failure of the operator to align the AAC power source was also examined. This failure scenario will increase the SBO CDF frequency approximately by a factor of five. Improving the operator ability by a factor of ten, the SBO CDF will be reduced about four times.

As shown in FIG 3 the Natural Circulation System has four solenoid valves. The failure of these valves contributes with 93% to the probability of the NCR failure as shown in Table VII. This is an indication of the importance of having a strict inspection and maintenance program to keep these valves within the required performance along their lifetime.

The SBO CDF of the LABGENE reactor was compared to those presented in NUREG/CR-6890[5]. This is shown in FIG. 9. It can be argued that quantitative comparisons made among different studies can be misleading. In fact, different methodologies can be employed and operational and design characteristics are plant specific. For the LABGENE reactor, however, the two design configurations – with and without the NRC system- are compared using the same basis.

In conclusion, the LABGENE reactor plant under construction in Brazil has been designed to ensure adequate protection against station blackout. This can be achieved by having sufficient AC and DC power sources capacity and by also having passive natural convection system to remove the residual heat from the reaction core.

Table 6 – Partial list of cut sets of NCR core residual heat removal system fault tree

Cut No.	% Total	% Cut Set	Frequency	Cut Sets
1	23.3	23.3	5.000E-004	F1-2-1-VALV-VP1
2	46.6	23.3	5.000E-004	F1-2-2-VALV-VP2
3	69.9	23.3	5.000E-004	F1-2-3-VALV-VS1
4	93.2	23.3	5.000E-004	F1-2-4-VALV-VS2
5	97.4	4.2	9.000E-005	F1-3-2-VALV-AUTO-OPEN, F1-3-1-VALV-MAN-OPEN
6	99.3	1.9	4.000E-005	F1-5-VALV-CABLE
7	99.7	0.5	9.600E-006	F1-4-VALV-PN-DC
8	100.0	0.4	8.000E-006	BATTERIES-A&B-MCO

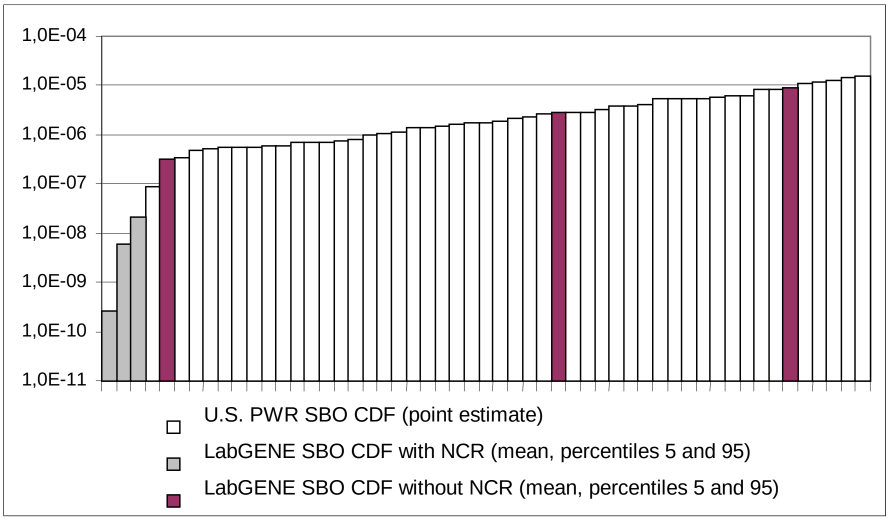


Fig. 9. SBO CDF of LABGENE reactor and USA PWR NPPs

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