Passive safety of new designs of nuclear power plants

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This study describes RELAP5 computer code for thermal-hydraulic analysis of a typical Pressurized Water Reactor (PWR). RELAP5 is used to calculate the thermal-hydraulic characteristics of the reactor core and the primary loop under steady-state and hypothetical accidents conditions. New designs of nuclear power plants are directed to increase safety by many methods like reducing the dependence on active parts, such as safety pumps, fans, and diesel generators, and replacing them with passive features such as gravity draining of cooling water from tanks, and natural circulation of water and air. In this work, high and medium pressure injection pumps are replaced by passive injection components. Different break sizes in cold leg pipe are simulated to analyze to what degree the plant is safe, without any operator action, by using only these passive components.

هذا البحث يصف نموذج محاكاة كامل بواسطة كود RELAP5 و المخصص لتحليل الأداء الحراري لمفاعلات الطاقة خاصة مفاعلات الماء المضغوط. كود RELAP5 استخدم في هذا البحث لدراسة الأداء الحراري لقلب مفاعل ماء مضغوط و كذلك الأداء الحراري للدائرة الابتدائية للمبرد في الحالات التالية: ١- حالة التشغيل العادي ٢- حالات حوادث افتراضية. هذا البحث يعرض إلى أى مدى يكون مفاعل الماء المضغوط المستخدم آمنا مع استخدام أنظمة أمان انفعالية Passive systems لا تعتمد على أجزاء ميكانيكية (مثل المضخات أو التربينات أو مولدات الديزل) مطلقا والاستعاضة عن ذلك باستخدام أنظمة أمان انفعالية تعتمد على خصائص طبيعية لا يمكن تعطيلها بأى شكل من الأشكال مثل الاعتماد على ضخ الماء بالجاذبية الأرضية أو التبريد بحمل الحراري الطبيعي دون استخدام أى أجزاء ميكانيكية. في هذا البحث تم استبدال مضخات الضغط العالي والضغط المتوسط في نظام الطوارئ للتبريد ECCS بنظامين انفعاليين ودراسة حوادث افتراضية مختلفة لمعرفة إلى أى مدى يكون النظام آمنا باستخدام هذه الأنظمة دون أى تدخل للمشغل.

Keywords: Simulation, PWR, Emergency Core Cooling System (ECCS), Passive plant, RELAP5

1. Introduction

A major safety advantage of passive plants is that long-term accident mitigation is maintained without operator action or reliance on off-site or on-site AC power. New passive plants use extensively analyzed and tested passive systems to improve the safety of the plant. The passive safety systems are significantly simpler than traditional PWR safety systems and do not require the large network of safety support systems needed by typical nuclear plants. That includes AC power, heating, ventilation, air conditioning, cooling water systems, and the seismic buildings needed to house these components [1].

Passive systems use gravity, natural circulation and compressed gas. No pumps, fans, diesels, chillers, or other rotating machines are used in the safety sub-systems. New passive plants designs have fewer valves, less piping, less control cable, fewer pumps and less seismic building volume than a similarly sized conventional plant [2]. Fig. 1 shows a comparison between current typical Pressurized Water Reactor (PWR) designs and passive designs (AP1000 as an example).

The simplified construction will also reduce operator actions. The passive design means operators would not need to take immediate action after an accident, with the reactor, instead, safely shutting down on its own. Also, with passive safety features and extensive plant simplifications that enhance the construction, operation, maintenance and safety.

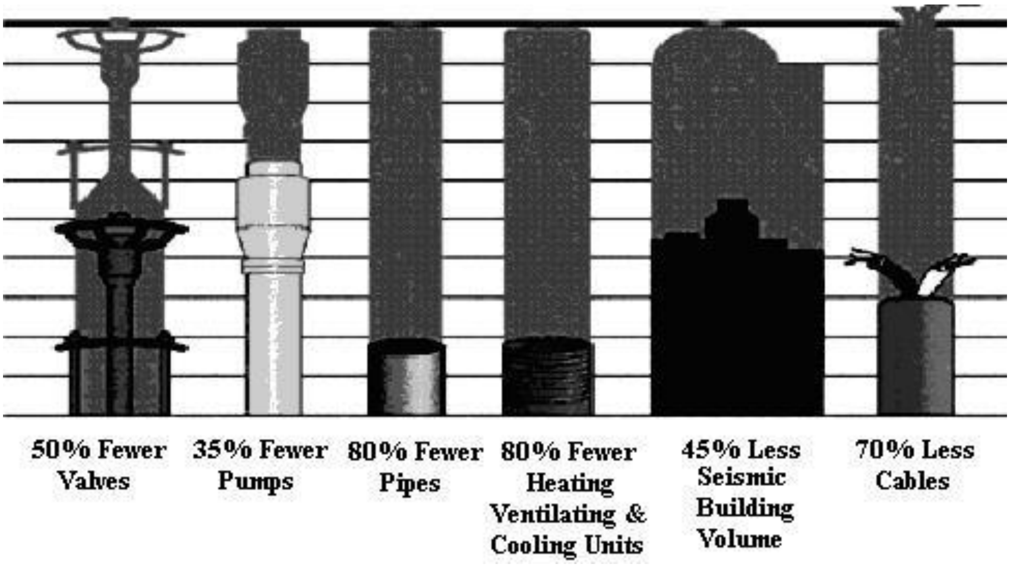


Fig. 1. Comparison between Active current designs and Passive designs (AP1000) of PWR.

2. RELAP5 code

The RELAP5 hydrodynamic model is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. It was developed at the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission (NRC). Code uses include analyses required to support rulemaking, licensing audit calculations, evaluation of accident mitigation strategies, evaluation of operator guidelines, and experiment planning analysis. The code has been developed and used for the analysis of light water reactors (and also for CANDU analyses) with a loop design. Although the RELAP code has been extensively used in the analyses of light water reactors, and has also been used in the transient analyses of advanced Westinghouse passive plants, the introduction of a new reactor and supporting systems poses great challenges to the development of an appropriate plant representation in RELAP [3].

3. Description of the model

The reference plant chosen for the present study is a three-loop typical PWR design (modeling of two- and four-loop designs is similar). The major components of the plant are:

1. Pressurizer (1 for the plant),
2. Steam generator (1 for each loop),
3. Reactor pressure vessel (1 for the plant),
4. Reactor coolant pump (1 for each loop),
5. Connecting pipes, and
6. Passive safety injection system (2 systems)

The nominal plant operating parameters are given in table 1.

The nodalization of the reference PWR plant model used in this thesis represents the standard nodalization scheme used at the Idaho National Laboratory (INL). Fig 2 shows the two passive safety injection systems used in this work: core makeup tank and accumulator. These two passive injection systems are:

1. High pressure safety injection (Core makeup tanks CMTs)
2. Medium pressure safety injection (Accumulator).

Table 1
Initial conditions for the reference plant

Parameter	Value
Reactor power (MWth)	2300
Coolant pressure (MPa)	15.51
SG pressure (MPa)	5.5
Active core height (m)	4.1
Core flow rate (kg/s)	12,725
Inlet core temperature (K)	559
Outlet core temperature (K)	592

High pressure safety injection CMTs used when the normal makeup system is inadequate or is unavailable. CMTs filled with cold borated water and designed to function at any Reactor Coolant System (RCS) pressure using only gravity, and the temperature and height differences from the reactor coolant system cold leg as the motivating forces [4]. These tanks are located above the RCS loop piping. If the water level or pressure in the pressurizer reaches a set low level, the reactor, as well as the reactor coolant pumps, is tripped and the CMT discharge isolation valves open automatically. The water from the CMTs recirculates then flows by gravity through the reactor vessel. It is always at primary pressure and natural circulation is established when valves are open and cold borated water enters reactor and hot primary water flows to CMT head. The detailed equations of the CMTs founded in both refs. [5 and 6].

Medium pressure safety injection Accumulators are required for Loss Of Coolant Accidents (LOCAs) to meet the immediate need for higher initial makeup flows to refill the reactor vessel lower plenum and downcomer following RCS blow down. The accumulators are pressurized to about 700 psig with nitrogen gas [7]. The pressure differential between the pressurized accumulators and the dropping RCS pressure ultimately forces open check valves that normally isolate the accumulators from the RCS. The accumulators continue delivery to supplement the CMTs in maintaining water coverage of the core.

4. Model validation

As the results issued from numerical simulations of transients and accidents in real reactors cannot be compared with theoretical or experimental results, the detection of a mistake in the numerical model is impossible. Thus a preliminary work of validation of the model is compulsory [8]. The validation is obtained by performing a computation under normal operating conditions. The transients extremity remains clogged. As computations are initialized approximately in operating conditions, the convergence of the results around the initial conditions is sufficient to prove that the model is correct and fig. 3 shows the convergence of the results of the pressure of the pressurizer when using different initial conditions.

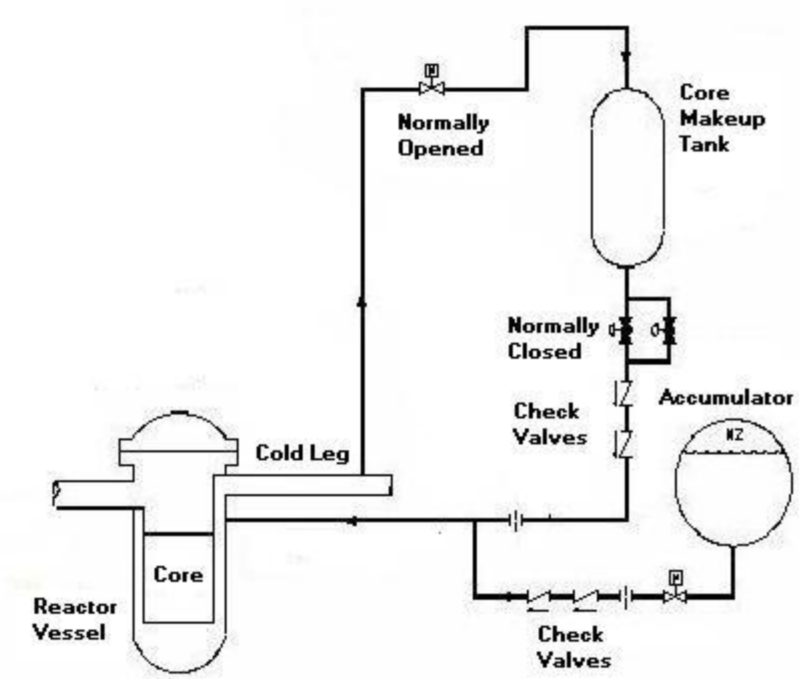


Fig. 2. Two Passive Injection systems used in the model.

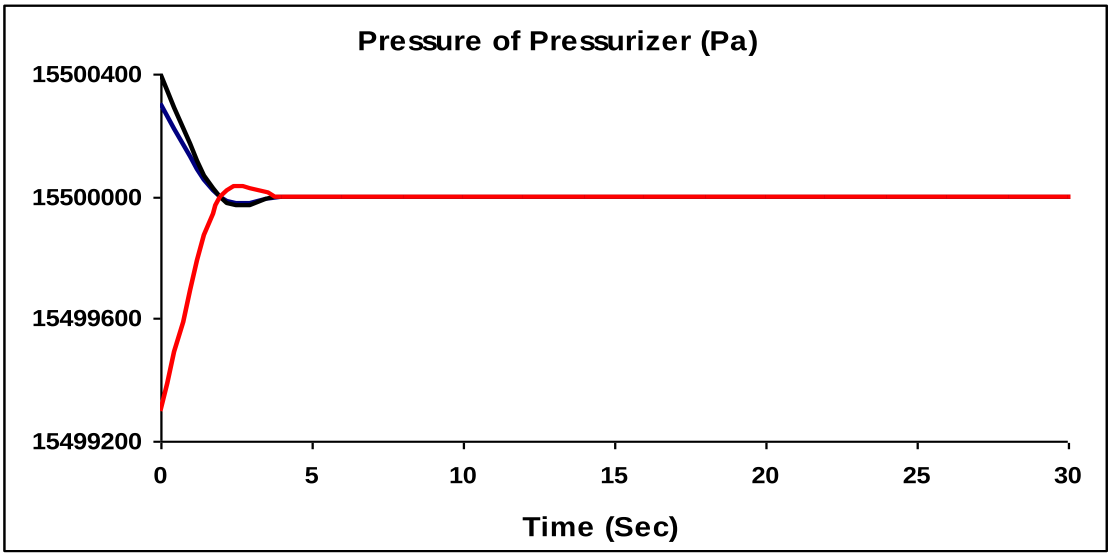


Fig. 3. Convergence of the coolant system pressure around operating condition (15.5 MPa) for different initial conditions.

5. Results and discussion

After running the model under normal operation conditions for suitable time to reach to stability and after checking of the model consistency by changing the initial conditions fig. 3 as shown in previous section, the transients and accidents simulation may be now carried out.

Small Break Loss Of Coolant Accident (SBLOCA) is taken as the accidental base case in this work. It is modeled in RELAP5 by simulating the rupture in the model by using an imaginary valve which modeled between the place of the break and a sink volume. The volumes of both CMT and accumulator taken from refes. [9 and 10].

The base case of SBLOCA sequence assumes that the small breaks with different diameters (starting from 5 inches break) occur in the cold leg of the plant. The recovery action of the two used passive emergency core cooling system is examined in all the cases. The main attention is “Can we say the plant is safe by using these two passive ECCS in the case of SBLOCA for 1500 seconds after the accident?” and if the answer is yes, “To which size we can say that?”.

Table 2. shows the analytical results of both of the base case (without action) and the mitigated case (with passive emergency core cooling) of the following accident scenarios: 5-8 inches rupture in cold leg of the plant for 1500 seconds after the accidents.

Figs. 4 and 5 show for all the cases the following:

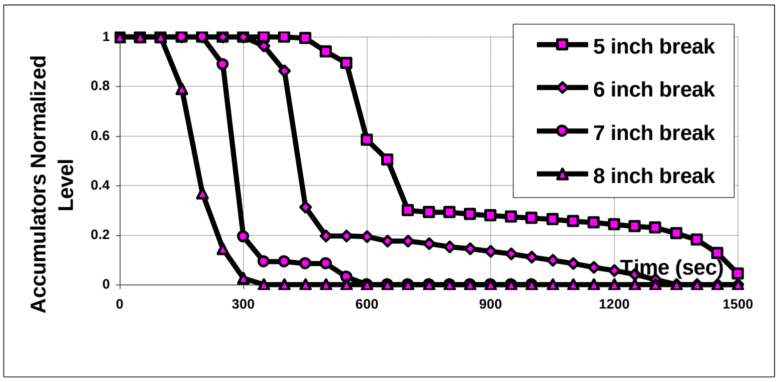
1. Normalized level of both passive ECCS used.
2. Level of water above the bottom of the core, and
3. Maximum clad temperature (which must not exceeds 1472 k [11]).

6. Conclusions and future work

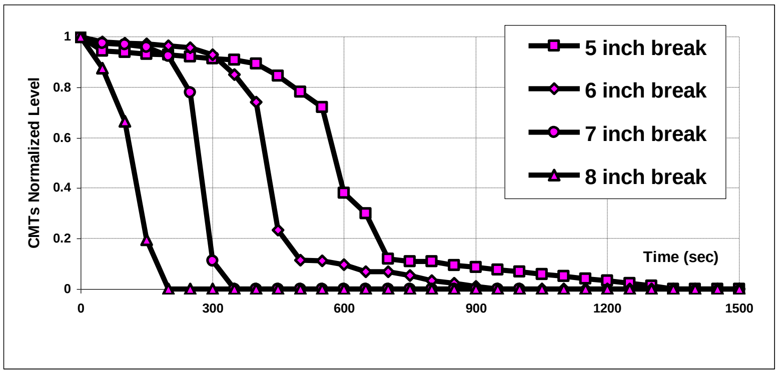
It is clear after the accidental analysis of SBLOCA of the typical pressurized water reactor with only passive ECCS: HPI passive tanks and MPI accumulators. Figs. 2 to 5 clearly showed that the new designs of nuclear power plants (PWR) that use only passive ECCS, we can say “the current plant is safe in the case of SBLOCA until size of 7 cm break 2000 sec after the accident by using only two short term passive ECCS without any operator action”.

Table 2
SBLOCA analytical results in cold leg and mitigation measure (sequences / sec) for different break sizes

	5 inch break		6 inch break		7 inch break		8 inch break	
Progression	Base Case	Mitigation measure	Base Case	Mitigation measure	Base Case	Mitigation measure	Base Case	Mitigation measure
Simulation starts	0	0	0	0	0	0	0	0
Accident begins	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Reactor trip	15	15	15	15	15	15	15	15
Core begins to uncover	180	220	95	95	67	75	26	28
HPI starts	----	38	----	25	----	20	----	17
Accumulator starts	----	620	----	300	----	200	----	100
HPI empty	----	1500	----	950	----	350	----	200
Accumulator empty	----	----	----	1350	----	600	----	350
Core completely uncover	1200	----	600	----	500	----	100	1400
Max clad temperature › 1472 k	1200	----	720	----	630	----	400	1500
Simulation ends	1500	1500	1500	1500	1500	1500	1500	1500
Consequence of accident and mitigation measure	Core melt	Core melt is prevented	Core melt	Core melt is prevented	Core melt	Core melt is prevented	Core melt	Core melt

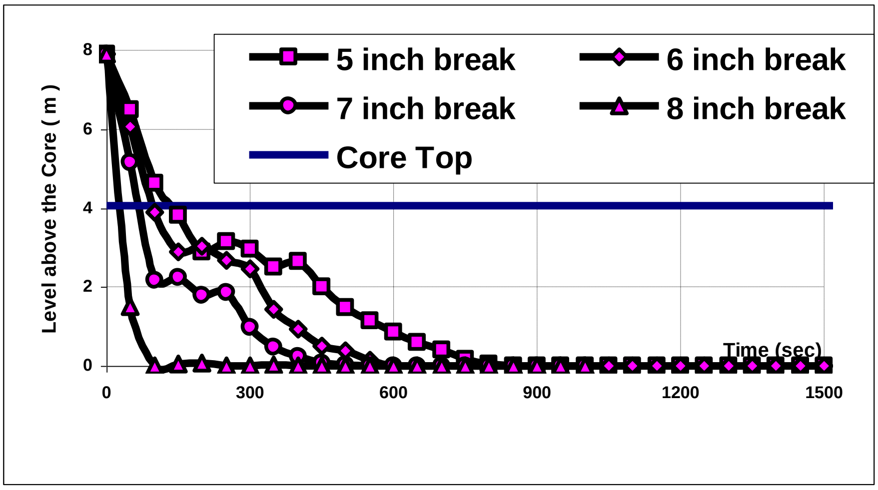


(a)



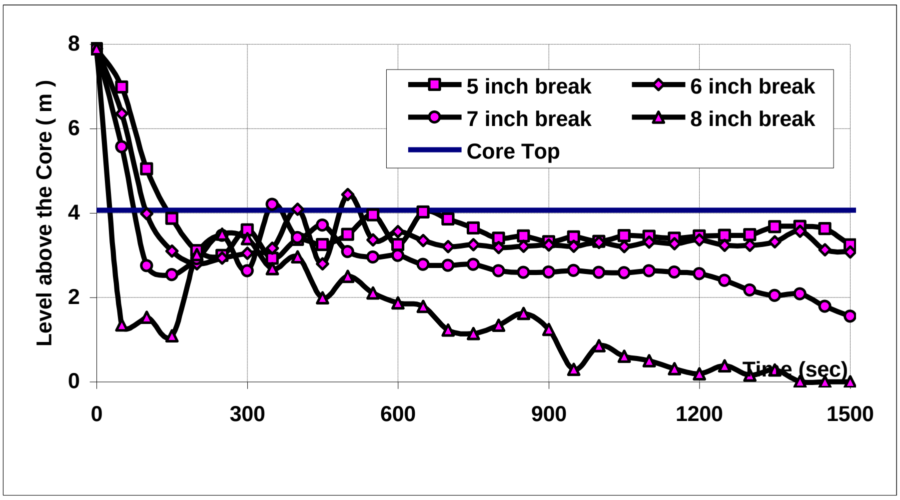
(b)

Fig. 4. Normalized level of the two passive ECCS used: (a) Accumulators. (b) CMTs. 1500 sec after the accident for different cold leg pipe break sizes.



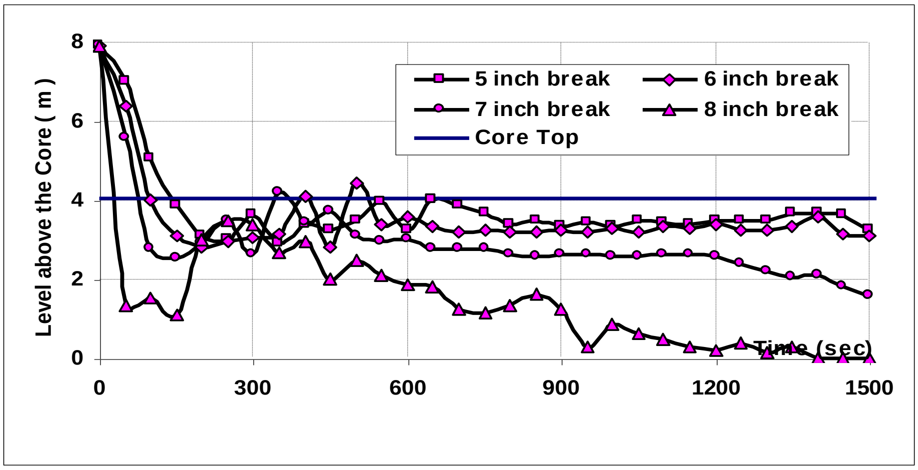
Level above the Core (m)
8
6
4
2
0
0
300
600
900
1200
1500
Time (sec)
5 inch break
6 inch break
7 inch break
8 inch break
Core Top

(a)



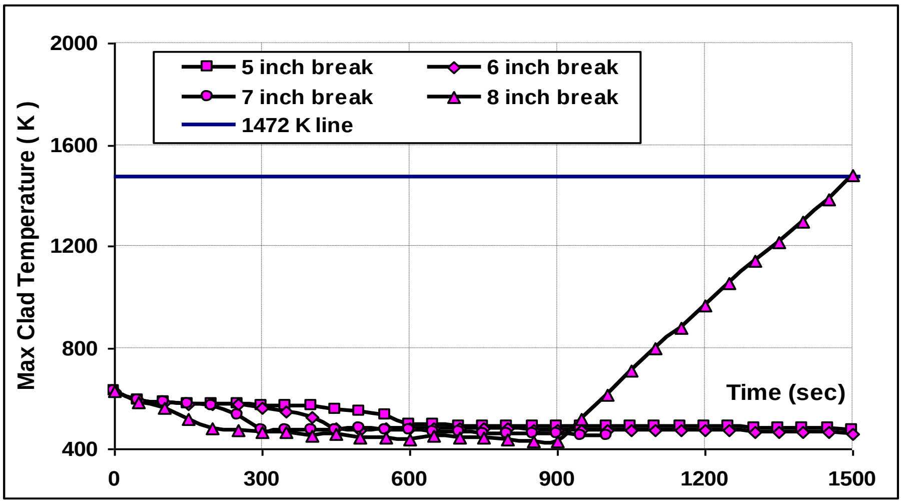
Level above the Core (m)
8
6
4
2
0
0
300
600
900
1200
1500
Time (sec)
5 inch break
6 inch break
7 inch break
8 inch break
Core Top

(b)



Level above the Core (m)
8
6
4
2
0
0
300
600
900
1200
1500
Time (sec)
5 inch break
6 inch break
7 inch break
8 inch break
Core Top

(c)



(d)

Fig. 5. Time response (1500 sec after the SBLOCA occurs) for different cold leg pipe break sizes of: (a) Level above the core (without action). (b) Level above the core (with passive ECCS). (c) Maximum clad temperature (without action). (d) Maximum clad temperature (with passive ECCS).

Also we can say that the new designs enhance "safety margins" of nuclear power plants by using more passive safety systems because the very quick response of the passive systems which do not depend on operator action. That is because the very important conclusion that the new designs succeeded in minimize the dependency on the operator action in some kinds of accidents as SBLOCA and SBO.

The future work must be concentrated on the analysis of the other different types of accidents to see to which degree the using of passive safety systems increase the safety margins and also modeling other passive safety systems to cover larger time scales after the accidents.

Nomenclature

CMT	Core makeup tank,
ECCS	Emergency Core Cooling System,
INEL	Idaho National Engineering Laboratory,
LOCA	Loss Of Coolant Accidents,
NRC	Nuclear Regulatory Commission,
PWR	Pressurized water reactor,
RCS	Reactor Coolant System,
SBLOCA	Small Break Loss of Coolant Accident, and
SBO	Station blackout.

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