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SUBSTANTIATION OF MODERNIZED BLACKOUT & LOSS-OF-COOLANT ACCIDENT MANAGEMENT STRATEGY AT NUCLEAR POWER PLANTS WITH WWER

В.І. Скалозубов, В.М. Спінов, Д.В. Спінов, Т.В. Габлая, В.Ю. Кочнєва, Ю.О. Комаров. **Обґрунтування модернізованої стратегії управління аваріями з повним тривалим знеструмленням при течах реакторного контуру ядерних енергоустановок з реакторами ВВЕР.** Проведений аналіз відомих результатів розрахункового моделювання кодом RELAP5/V.3.2 аварій з повним тривалим знеструмленням та течами другого контуру ядерних енергоустановок з ВВЕР показав, що проектні стратегії управління такими аваріями проектними пасивними системами безпеки не забезпечують необхідні умови безпеки щодо максимально допустимої температури оболонок твелів, щодо мінімально допустимого рівня теплоносія в реакторі і живильної води в парогенераторах. Розроблено консервативну теплогідродинамічну модель проектної і модернізованої стратегії управління аваріями з течами реакторного контуру та повним тривалим знеструмленням ядерної енергоустановки з ВВЕР. Проектна стратегія управління аваріями здійснюється проектними пасивними системами безпеки: запобіжними клапанами систем компенсації тиску і пароскидальних пристроїв 2-го контуру, а також гідроємностями системи аварійного охолодження активної зони реактора. Модернізована стратегія управління аваріями з течами реакторного контуру та повним тривалим знеструмленням ядерної енергоустановки з ВВЕР. Проектна стратегія здійснюється перспективними системами пасивного відводу тепла від активної зони реактора і підтримки рівня теплоносія в реакторі та живильної води в парогенераторах. Основні консервативні допущення представленої моделі аварій з течами реакторного контуру та повним тривалим знеструмленням: повна тривала відмова усіх електронасосів активних систем безпеки; температура ядерного палива в центральній частині паливної матриці твела приймається максимально допустимою; не враховується вплив на аварійний процес витрати «вибігу» турбоживильного насоса та рівня теплоносія в компенсаторі тиску. В результаті розрахункового моделювання встановлено, що при проектній стратегії управління аваріями з течами реакторного контуру та повним тривалим знеструмленням порушення умов безпеки визначені для всього діапазону розмірів теч. При модернізованій стратегії управління аваріями умови безпеки забезпечені протягом 72 годин аварійного процесу та більше. Представлені результати розрахункового моделювання стратегій управління аваріями з повним тривалим знеструмленням ядерних енергоустановок можуть бути використані для модернізації і вдосконалення симптомно-орієнтрованих аварійних інструкцій та посібників з управління важкими аваріями на ядерних енергоустановках із реакторами типа ВВЕР. Застосування отриманих результатів розрахункового моделювання стратегій управління аваріями з повним тривалим знеструмленням у загальному випадку не обгрунтовано для інших типів реакторної установки. У цьому випадку необхідна розробка розрахункових моделей управління аваріями з повним тривалим знеструмленням, що враховують специфіку конструкційно-технічних характеристик та умов експлуатації систем, важливих для безпеки ядерних енергоустановок.
Ключові слова: стратегія управління аварією, знеструмлення, теча реакторного контуру

V. Skalozubov, V. Spinov, D. Spinov, T. Gablaya, V. Kochnyeva, Yu. Komarov. **Substantiation of Modernized Blackout & Loss-of-Coolant Accident Management Strategy at Nuclear Power Plants with WWER.** The analysis of the known results of RELAP5/V.3.2 simulation for loss of coolant & blackout accidents at WWER nuclear power plants showed that the design accident management strategies with design passive safety systems do not provide the necessary safety conditions for the maximum permissible temperature of fuel claddings, the minimum permissible level of coolant in the reactor and feed water in the steam generators. A conservative thermohydrodynamic model for a design and modernized blackout & loss-of-coolant accident management strategy at a nuclear power plant with WWER has been developed. Design passive safety systems carry out the design accident management strategy: pressurizer safety valves, secondary steam relief valves, and hydraulic reservoirs of the emergency core cooling system of the reactor. Promising afterheat removal passive systems and the reactor level and steam generator water level control systems carry out the modernized blackout & loss-of-coolant accident management strategy. The main conservative assumptions of the presented model of blackout & loss-of-coolant accidents: complete long-term failure of all electric pumps of active safety systems, the temperature of nuclear fuel in the central part of the fuel matrix is assumed as the maximum allowable one, effect of "run down" flow of a turbine feed pump and the coolant level in pressurizer on accident process is not considered. Computational modelling has found that violations of the safety conditions are over the entire range of leak sizes for the design blackout &

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loss-of-coolant accident management strategy. For the modernized blackout & loss-of-coolant accident management strategy, safety conditions are provided for 72 hours of the accident and more. The presented results of computational modelling of blackout accident management strategies for nuclear power plants can be used to modernize and improve symptom-informed emergency instructions and guidelines for the severe accident management at nuclear power plants with WWER. Application of the results of computational modelling of blackout accident management strategies is generally not substantiated for other types of reactor facilities. In this case, it is necessary to develop calculated models for blackout accident management taking into account the specifics of the structural and technical characteristics and operating conditions for safety related systems of nuclear power plants.

Keywords: accident management strategy, blackout, loss-of-coolant accident

Introduction

Improvement of blackout accident management strategy for nuclear power facilities (NPF), and modelling and analysing of combined initial accident events with multiple failures of the safety related systems is one of the main lessons of Fukushima-Daiichi accident in 2011.

Design blackout accident management strategies for NPF with WWER come actually to the staff actions to recover power supply, to decrease the reactor and steam generator pressure, and to connect low-head cooling systems. Such approach has the following shortcomings according to Fukushima-Daiichi lessons.

1. Recovery of power supply of the power unit can be impossible as for the objective reasons because of accident conditions and for the subjective reasons because of inadvertent wrong staff actions in stressful situations.

2. Preventive pressure decrease in NPF loops can have also negative effects. For example, preventive pressure decrease in containment of the Fukushima-Daiichi-1 led to destructive explosions.

After Fukushima-Daiichi accident, nuclear powers (including Ukraine) came to a conclusion about expediency of modernization of blackout accident management strategies. Alternative passive (without power supply) safety systems are developed to modernize these strategies: systems of after-heat removal passive with natural circulation and steam-driven emergency pumps.

Implementation of such alternative passive systems demands to substantiate their efficiency for blackout accident management. Substantiation of efficiency of the modernized blackout accident management strategy can be based on results of calculated modelling of accidents. It defines relevance of the offered work.

Analysis of recent publications and problem statement

The work [1] considers the safety of the NPP with WWER and modelling of possible accidents. However, modelling of blackout accidents are not considered.

The work [2] generalizes results of modelling of accidents at different NPFs. Modelling of blackout accidents is also not considered.

The work [3] considers verification of calculated thermohydrodynamic codes for conditions of NPF with WWER-1000. Verification of codes for blackout accidents is not considered.

The work [4] presents results of modelling of loss-of-coolant accidents in WWER with the ATHLET Code. Modelling of blackout accidents is not considered.

The work [5] presents the overview of results of accident modelling at the USA NPFs. However, results of modelling of blackout accidents are also not provided.

The work [6] presents results of modelling of loss-of-coolant accidents at NPF with WWER-1000 to substantiate diversification of nuclear fuel. Modelling of blackout accidents is also not considered in this work.

The work [7] provides results of blackout accident modelling with the RELAP5 Code. It was revealed that safety conditions as to the maximum admissible temperature of fuel claddings are unambiguously violated during blackout accident. Results of this work defined relevance of further improvement of blackout accident management strategies for NPF.

The work [8] defines the general problems of forming of accident management strategies. However, forming of effective accident management strategies is not considered.

The work [9] provides loss-of-coolant accident management strategies. However, forming of effective blackout accident management strategies is also not considered.

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The work [10] proposes methods for modelling the conditions for water hammers during accidents in a pressurizer. However, these methods are not substantiated for blackout accident conditions.

The work [11] analyses the conditions for water hammers in active safety systems with electric pumps. However, the results of this work are not substantiated for passive safety systems in the blackout accident conditions.

The work [12] determines the conditions for water hammers in the transonic two-phase flows in the NPP armature. However, the results of this work do not cover the conditions for water hammers in other heat engineering equipment during blackout accidents.

The work [13] proposes an original approach to blackout accident management using a steam-driven emergency feed pump. However, the conditions for the effectiveness of such a pump for blackout accident management are not defined.

Purpose and objectives of the study

The purpose of the work is calculated and analytical substantiation of the modernized blackout accident management strategy for nuclear power facilities with WWER.

The work tasks.

1. To develop conservative thermohydrodynamic model of blackout & loss-of-coolant accident (LOCA).

2. To analyse the results of calculated modelling and to substantiate efficiency of the modernized management strategy for loss-of-coolant & blackout accident.

Conservative thermohydrodynamic model of loss-of-coolant & **blackout accident.** The conservative thermohydrodynamic model of loss-of-coolant & blackout accident is considered for two accident management strategies:

– the design accident management strategy (DAMS) with design passive safety systems;

– the modernized accident management strategy (MAMS) with design passive safety systems and prospective passive safety systems:

– the afterheat removal passive & level control system of the reactor – ARPS R;

– the afterheat removal passive & level control system of the steam generator – ARPS SG.

ARPS R includes two subsystems:

– the afterheat removal passive subsystem of the reactor with steam-driven emergency pump (SDEP) – ARPS R1;

– the afterheat removal passive subsystem of the reactor with natural circulation – ARPS R2.

ARPS SG includes two subsystems:

– a subsystem with the SG steam-driven auxiliary feedwater pump (SDAFP) – ARPS SG1;

– a subsystem with the SG closed circuit of natural circulation – ARPS SG2.

Main conservative assumptions.

1. Complete failure of all electric pumps of a safety system, and impossibility of restoration of auxiliary supply of the power unit within not less than 72 hours from the initiation of accident are modelled.

2. Nuclear fuel temperature T_F in the central part of a fuel matrix is accepted as the maximum admissible one for nuclear safety conditions (2800 °C).

3. Accident effect of “run down” flow of a turbine feed pump and the coolant level in the pressurizer is not considered.

The mass and heat balance equations for the reactor volume, “free” of structures, V_R:

$$\frac{\mathrm{d}(\rho_v V_{vR})}{\mathrm{d}t} = G_{Tv} - G_{v1R} - G_{2R}; \qquad V_R = V_{vR} + V_T, \tag{1}$$

$$\rho_T \frac{dV_T}{dt} = G_K + G_{gp} + G_{1R} + G_{2R} - G_{Tv} - G_{LOC} + G_{ge}, \tag{2}$$

$$\frac{d(\rho_v V_{vR} i_v)}{dt} = G_{Tv} r_v - (G_{1R} + G_{2R}) i_v, \tag{3}$$

$$\rho_T \frac{d(V_T i_T)}{dt} = (G_K + G_{gp} - G_{LOC}) i_T + G_{1R} i_{1R} + G_{2R} i_{2R} - G_{Tv} r_v + G_{ge} i_{ge} \,. \tag{4}$$

The heat balance equations for system “reactor – SG” and ARPS R2:

$$N(t) = F_{.1} R_T^{-1} (T_F - T_0) + F_2 R_{\mathrm{VG}}^{-1} (T_0 - T_1) + G_{LOC} i_T \,, \tag{5}$$

$$G_{2R} r_c = \alpha_0 F_{.2R} (T_v - T_{os}) - C_p G_{2R} T_{cs} - T_{2R}) \,. \tag{6}$$

Motion equations as a quasistationary approximation for systems of a primary loop, a steam drive of SDEP and ARPS R2:

$$G_K = \Pi_T \sqrt{\rho_T (\rho_v - \rho_T) g h_{1K} / \xi_{1K}} \,, \tag{7}$$

$$G_{v1R} = \mu_v \Pi_{vR} \sqrt{2\rho_v (P_R - P_{SA})} \,, \tag{8}$$

$$G_{2R} = \Pi_{2R} \sqrt{\rho_T (\rho_v - \rho_T) g h_{2R} / \xi_{2R}} \,. \tag{9}$$

$$G_{ge} = \begin{cases} \mu_{ge} \Pi_{ge} \sqrt{2\rho_T (P_{ge} - P_{vR})}, \; P_{ge} > P_{vR}, \; t < t_{ge}; \\ 0, \quad P_{ge} \le P_{vR}, \quad t \ge t_{ge}. \end{cases} \tag{10}$$

Coolant loss flow as a quasistationary approximation:

$$G_{LOC} = \begin{cases} \mu_{LOC} F_{LOC} \sqrt{2\rho_T (P_R - P_{os})}, \; G_{LOC} < G_{cr}; \\ G_{cr}, \; G_{LOC} \ge G_{cr}, \end{cases} \tag{11}$$

$$G_{1R} = \begin{cases} G_{SR}, \quad P_R > 0.3 \text{ MPa}; \\ 0, \; P_R \le 0.3 \text{ MPa}, \end{cases} \tag{12}$$

where ρ_v, ρ_T is a steam and the coolant density, respectively, V_{vR}, V_T is a steam and the coolant volume in the reactor, respectively, t *is* time, G_{Tv}, G_{v1R}, G_{2R}, G_K, G_{gp} is a mass flow rate of steam generation intensity, steam in a SDEP steam drive, in ARPS R2, primary natural circulation, “run down” of the main coolant pump, respectively, G_{SR} is the maximum output of electric pumps of emergency core cooling systems of the reactor (ECCS), G_{cr} is a critical loss in the transonic modes of a two-phase flow, P_R is the reactor pressure, i_v, i_T, i_{1R}, i_{2R} is a specific enthalpy of steam, the coolant in the reactor, in ARPS R1 and ARPS R2, respectively, r_v, r_c is the latent heat of steam generation and condensation, respectively, T_o, T_l, T_v, T_{os}, T_{2R} is temperature of fuel cladding, feedwater in SG, steam, environment in containment, the coolant at the outlet of ARPS R2, respectively, $N(t)$ is the afterheat power, F_1, F_2 is the total heat-transfer surface area in a nuclear core and SG, respectively, α_0 is heat-transfer coefficient on ARPS R2 surface, F_{2R}, h_{2R} is a heat-transfer surface area and height of ARPS R2, respectively, C_p is the specific heat capacity of condensate, T_{Cs} is condensate saturation temperature, g *is* acceleration due to gravity, Π_T, Π_{2R}, Π_{vR} is the throat area of the primary loop, ARPS R2 and the ARPS R1 steam drive, respectively, h_{1K} is the primary leveling height, P_{SA} is pressure in hydraulic reservoirs of ECCS, ξ_{1K}, ξ_{2R} is total coefficient of hydraulic resistance of the primary loop and ARPS R2, respectively, μ_{LOC}, μ_v is a flow coefficient of a steam drive and the loss, respectively, F_{LOC} is the equivalent throat area of a loss, μ_{ge}, Π_{ge}, P_{ge}, t_{ge}, i_{ge} is a flow coefficient, the throat area, pressure, outflow time and a specific enthalpy of the coolant in hydraulic reservoirs of ECCS, respectively.

Coefficients of thermal resistance of fuel element and interloop volume:

$$R_{f0} = \delta_F / \lambda_F + \delta_g / \lambda_g + \delta_0 / \lambda_0 \,,$$

$$R_{SG} = 1/\alpha_1 + \delta_{\mathrm{SG}} / \lambda_{\mathrm{SG}} + 1/\alpha_2 \,,$$

where δ_F, δ_g, δ_0, δ_{SG} is thickness of a fuel matrix, a gas gap, a fuel cladding and heat-exchange tubes of SG, respectively, λ_F, λ_g, λ_0, λ_{SG} is a thermal conductivity of a fuel matrix, a gas gap, a fuel cladding

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and heat-exchange tubes of SG, respectively; α_1, α_2 is heat-transfer coefficient in a nuclear core and on a pipe surface of SG, respectively.

The mass and heat balance equations for SG volume, "free" of structures, V_{SG}:

$$\frac{\mathrm{d}(\rho_v V_v)}{\mathrm{d}t} = G_{iv} - G_{v1g} - G_{2g}; \quad V_{SG} = V_v + V_l, \tag{13}$$

$$\rho_i \frac{\mathrm{d}V_l}{\mathrm{d}t} = G_{1g} + G_{2g} - G_{iv}, \tag{14}$$

$$\frac{\mathrm{d}(\rho_v V_v i_v)}{\mathrm{d}t} = G_{iv} r_v - (G_{v1g} + G_{2g}) i_v, \tag{15}$$

$$\rho_l \frac{\mathrm{d}(V_l i_l)}{\mathrm{d}t} = G_{1g} i_{1g} + G_{2g} i_{2g} - G_{lv} r_v. \tag{16}$$

The heat balance equations for ARPS SG2:

$$G_{2g} r_c = \alpha_0 F_{2g} (T_v - T_{os}) - C_p G_{2g} (T_{cs} - T_{2g}). \tag{17}$$

Motion equations as a quasistationary approximation for subsystems of SG ARPS:

$$G_{v1g} = \mu_v \Pi_{v1g} \sqrt{2\rho_v (P_{SG} - P_D)}, \tag{18}$$

$$G_{2g} = \Pi_{2g} \sqrt{\rho_l (\rho_v - \rho_l) g h_{2g} / \xi_{2g}}, \tag{19}$$

$$G_{1g} = \begin{cases} G_{AP}, \ P_{vg} > 0.3 \text{ MPa}; \\ 0, \ \ P_{vg} \le 0.3 \text{ MPa}, \end{cases} \tag{20}$$

where G_{iv}, G_{v1g}, G_{2g}, G_{1g} is a mass flow rate of steam generation intensity in SG volume, steam on a steam drive of ARPS SG1, steam in ARPS SG2, SDAFP of ARPS SG1, respectively, V_v, V_l is steam and feedwater volume in SG, , respectively, i_v, i_l, i_{ig}, i_{2g} is a specific enthalpy of steam and feedwater in SG, respectively, T_{2g} is condensate outlet temperature of ARPS SG2, Π_{v1g}, Π_{2g} is a throat area of a steam drive ARPS SG1 and ARPS SG2, respectively, G_{AP} is a rated output of the emergency feed water pump, P_D is the deaerator pressure, h_{2g} is height of ARPS SG2, ξ_{2g} is total coefficient of hydraulic resistance of ARPS SG2.

After transformations of combined equations (1) – (6), (13) – (17) taking into account compressibility of steam $d\rho_v / dt = (d\rho_v / dP) \cdot (dP / dt)$ we get the nonlinear combined equations:

$$\frac{dP_{vR}}{dt} = f_1[P_{vR}, V_T, i_T, P_{vg}, V_l, i_l, G_{LOC}, N(t)], \tag{21}$$

$$\frac{dV_t}{dt} = f_2[P_{vR}, V_T, i_T, P_{vg}, V_l, i_l, G_{LOC}, N(t)], \tag{22}$$

$$\frac{di_T}{dt} = f_3[P_{vR}, V_T, i_T, P_{vg}, V_l, i_l, G_{LOC}, N(t)], \tag{23}$$

$$\frac{dP_{vg}}{dt} = f_4[P_{vR}, V_T, i_T, P_{vg}, V_l, i_l, G_{LOC}, N(t)], \tag{24}$$

$$\frac{dV_l}{dt} = f_5[P_{vR}, V_T, i_T, P_{vg}, V_l, i_l, G_{LOC}, N(t)], \tag{25}$$

$$T_o = f_6[P_{vR}, V_T, i_T, P_{vg}, V_l, i_l, G_{LOC}, N(t)]. \tag{26}$$

Initial conditions for combined equations (21) – (26):

$$P_{vR}(t=0)=P_{vR_0};\ V_T(t=0)=V_{T_0};$$
$$i_T(t=0)=i_{T_0};\ P_{vg}(t=0)=P_{vg_0};$$
$$V_l(t=0)=V_{l_0};\ N(t=0)=N_0;\tag{27}$$
$$T_o(t=0)=T_{o_0}.$$

In the general case, combined equations (21) – (27) can be solved with a known Runge-Kutta method.

The analysis of the results

The main results of calculated modelling of combined equations (21) – (27) for the design and modernized loss-of-coolant & blackout accident management strategies (DAMS/MAMS) are given in Fig. 1.

Fig. 1 presents an calculated time of violation of safety conditions t_{cr} as to the maximum admissible temperature of fuel claddingts ($T_{\lim}$ = 1200 °C, $\mathbf{T}_0 = T_0/T_{\lim}$) and as to the level of complete drainage of the reactor core ($\mathbf{H}_{vR} = H_{Vr}/H_{vR0}$) for DAMS. Violation of safety conditions as to $\mathbf{T}_0 \geq 1$ and $\mathbf{H}_{Vr} = 0$ are determined for full range of primary loss $0 \leq \mathbf{F}_{LOC} = F_{LOC}/F_{gcT} \leq 1$ (F_{gcT} is the throat area of the primary main coolant pipeline).

Fig. 2 presents results of calculated modelling of maximum temperature of fuel claddings and the coolant level in the reactor for DAMS with a large loss-of-coolant accident (LLOCA) and blackout.

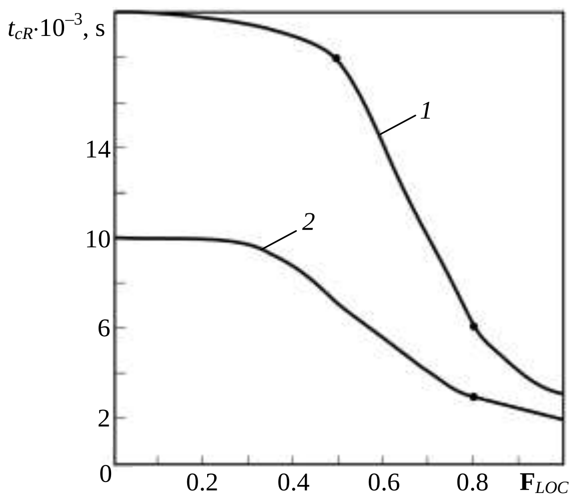

Fig. 1. An calculated time of initiation of violations of safety conditions t_{cr} for DAMS with LOCA and blackout: 1 – as to temperature of fuel cladding, 2 – as to drainage of the reactor core

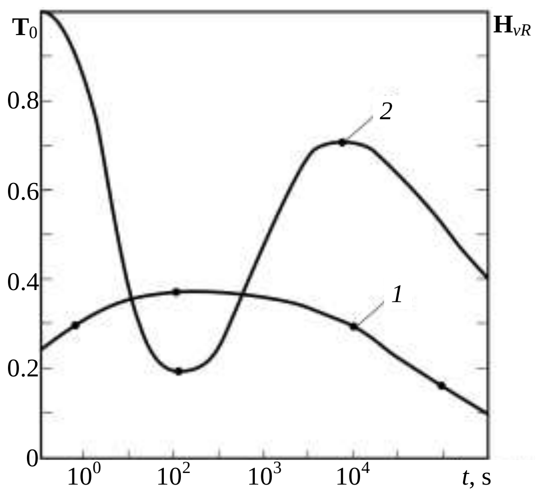

Fig. 2. Maximum temperature of fuel claddings and the coolant level in the reactor for MAMS with LLOCA and blackout: 1 – $\mathbf{T}_0=T_0/T_{lim}$, 2 – $\mathbf{H}_{vR}=H_{Vr}/H_{vR0}$

At the initial stage of accident (before 100 s), the coolant level in the reactor decreases, and temperature of fuel claddings increases. After 100 s of the accident, the maximum temperature of fuel elements claddings, and the coolant level increases because of effect ARPS R and ARPS SG. After 10^4 s of the accident, the coolant level in the reactor decreases because of shutdown of ARPS R1, but safety conditions as to temperature of fuel cladding and the coolant level in the reactor are provided for 72 h of the accident.

Conclusions

1. A conservative thermohydrodynamic model for a design and modernized blackout & loss-of-coolant accident management strategy at a nuclear power plant with WWER has been developed.

Design passive safety systems carry out the design accident management strategy: pressurizer safety valves, secondary steam relief valves, and hydraulic reservoirs of the emergency core cooling system of the reactor.

Promising afterheat removal passive systems and the reactor level and steam generator water level control systems carry out the modernized blackout & loss-of-coolant accident management strategy.

2. The main conservative assumptions of the presented model of blackout & loss-of-coolant accidents: complete long-term failure of all electric pumps of active safety systems, the temperature of nuclear fuel in the central part of the fuel matrix is assumed as the maximum allowable one, effect of "run down" flow of a turbine feed pump and the coolant level in pressurizer on accident process is not considered.

3. Computational modelling has found that violations of the safety conditions are over the entire range of leak sizes for the design blackout & loss-of-coolant accident management strategy. For the modernized blackout & loss-of-coolant accident management strategy, safety conditions are provided for 72 hours of the accident and more.

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