A Parameter Study of Large Fast Reactor Nuclear Explosion Accidents

J. R. Wiesel
A PARAMETER STUDY OF LARGE FAST REACTOR NUCLEAR EXPLOSION ACCIDENTS

J R Wiesel

ABSTRACT

An IBM-code EEM (Explosive Excursion Model) has been developed for calculating the energy releases associated with the explosive disassembly of a large fast reactor following a superprompt critical condition. The assumed failure chain of events and the possible core collapse following a fuel meltdown give the input data and initial conditions, the most important of which is the reactivity insertion rate at the moment of the explosive core disassembly. The dependence of the energy releases on the reactivity insertion rate, the Doppler reactivity feedback, the power form factor and the core size have been studied. The model enables a quick estimation of conservative values of the destructive mechanical energy releases following a nuclear explosion and gives suggestions as to how to reduce or even avoid such excursions.

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1. INTRODUCTION

The purpose of this report is to demonstrate the use of a calculation model for describing the parametric dependence of large energy releases associated with nuclear excursions leading to an explosive mechanical disassembly of the core. A range of ramp rates which should include those typical of a maximum credible and conceivable accident has been used as the input disturbance of the model. The effect of the core size and power flattening have been studied. The Doppler effect is incorporated in the model and the associated reactivity coefficient has been varied to take care of the uncertainties in determining this effect as well as the variations following from different core configurations and coolant void fractions.

2. THE NUCLEAR EXPLOSION ACCIDENT

There are two categories of accidents that could lead to core melting and a subsequent explosive mechanical expansion or disassembly of the core, namely primary reactivity accidents and loss-of-coolant accidents. The most important primary reactivity accidents are the fuel rod drop accident and the control or safety rod expulsion accident. For sodium-cooled reactors the most important loss-of-coolant accidents are the loss-of-coolant flow accident due to blockage or pump failure resulting in sodium voiding and the sodium voiding accident due to sodium superheat. Normally, a properly designed redundant safety system will terminate any power transient occurring. It is, however, valuable to assess the consequences of a severe accident if the protective systems fail. Such an analysis will give a design basis for the containment and make possible an economic evaluation of different protective measures, both internal core design measures and safety system measures. Generally the accidental nuclear excursion can be divided into three phases:

1. A combination of failures gives an initial power transient leading to damaged fuel elements - primary reactivity insertion.
2. The power transient is aggravated by a continuous core collapse leading to a more reactive core configuration - secondary reactivity insertion.
3. The power transient is terminated by an explosive disassembly of the core.

This report deals with phase 3 using the output from phases 1 and 2 as the initial conditions and input data. The dominant input parameter is the reactivity insertion rate or ramp reactivity. The ramp reactivity magnitude depends on the combination of the primary and secondary reactivity insertions following from the assumed accidental chain of events leading to core melt-down. The purpose of this report is neither to demonstrate the probability and consequences of different accidents nor to calculate the detailed core collapse following a fuel melt-down but only to study the disassembly phase of the excursion and give conservative values of the explosive energy releases during this phase.

3. THE CALCULATION MODEL

3.1 Introduction

An IBM-code EEM (Explosive Excursion Model) calculates the space-time phenomena associated with the explosive disassembly of a large fast reactor following a superprompt critical condition. The code is based on general equations describing the effects of a fast disassembly of the reactor core on the generated nuclear power with consideration of the fuel temperature reactivity feedback effect. The space-time behaviour of the nuclear characteristics is assumed separable and the time-dependence is described by one-point equations. Six delayed groups are used. The space-dependence is calculated using one-group diffusion theory. The code is developed for a one-zone, spherical, homogeneous model. The model is an adiabatic one and the nucleonics, thermodynamics and hydrodynamics are calculated in space and time. The power excursion is initiated by a linear reactivity insertion which generally brings the reactor into a prompt critical state. The main reactivity feedback is produced by the disassembly of the core material due to the internal pressure build-up in the fuel. This feedback depends on the core void and the equation of state of the vaporized fuel. When the voids are filled by melted and vaporized fuel, the pressure increases rapidly, thereby giving a fast separation of the fuel and a quickly in-
creasing negative expansion reactivity feedback. The influence of the Doppler effect, the reactor size, the nuclear power form factor and the reactivity ramp rate on the released nuclear energies have been studied.

The basic equations can be used to develop more sophisticated EEM codes, e.g. multizone codes, cylindrical, axial, pancake codes. Also, the effect of cladding strength can be taken care of as well as studies of different equations of state due to the fuel composition and the assumptions used. The effects of interfaces between different core zones are easily considered. For preliminary evaluations of orders of magnitudes of energy releases and relative parametric dependences the used model will give a clear and quick picture.

3.2 Assumptions

The assumptions for the general equations are:

a) space-time behaviour separable (neutron flux, power, energies)
b) one-point equations (nuclear time behaviour)
c) one-group diffusion theory (nuclear space dependence)
d) adiabatic heat model
e) first-order one-group perturbation theory (disassembly reactivity feedback, nearly time-constant disassembly internal reactivity density space-dependence)
f) nearly space-independent core density and nuclear constants
g) Doppler fuel temperature reactivity coefficient density varying with some power of the fuel temperature and weighted with some power of the neutron flux
h) source term neglected (nuclear equations)

The specific EEM-code assumptions are:

a) the geometrical model is a uniform, homogeneous sphere
b) the fuel pressure and temperature are functions of the internal energy density only
c) only the Doppler fuel temperature and the disassembly reactivity feedbacks are considered
d) the reactivity insertion is described by a ramp input
e) the pressure gradient is continuous at all surfaces including the external core surface
3.3 Nomenclature

t time
\( \tau \) prompt neutron lifetime
Q central internal energy density
\( k_p \) prompt reactivity
\( \lambda_i \) precursors decay constant
\( C_i \) precursors density variable
\( \beta_i \) delayed neutron fraction
k delayed reactivity
N space factor for internal energy density
E internal energy density
\( \varepsilon_{10} \) initial condition for delayed reactivity
\( \varepsilon_{20} \) initial drift rate for delayed reactivity
\( k_e \) external reactivity disturbance
\( k_{D} \) prompt fuel temperature reactivity feedback (Doppler effect)
\( k_{ex} \) disassembly reactivity feedback (mechanical expansion effect)
\( \varepsilon_1 \) step reactivity disturbance
\( \varepsilon_2 \) ramp reactivity disturbance
w internal reactivity density
V core volume
\( \sigma_D \) prompt fuel temperature internal reactivity coefficient density
(Doppler coefficient density)
T fuel temperature
\( \rho_f \) fuel mass density
\( \gamma \) prompt fuel temperature coefficient for fuel mass density
(thermal fuel expansion mass density coefficient)
\( \eta \) fuel mass density coefficient for internal reactivity density
r material space location
\( w_{ex} \) disassembly internal reactivity density
\( \gamma_{Do} \) initial central Doppler coefficient density
\( T_{ao} \) initial central fuel temperature
u power distribution exponent for Doppler coefficient density
(weight factor)
n temperature exponent for Doppler coefficient density
\( V_{E} \) time-variant core "disassembly" volume
Using the linear relationship between nuclear power and neutron density one obtains for the energy density in the centre of the core the following relations

\[ t \frac{d^2 Q}{dt^2} = k \frac{dQ}{dt} + \sum_{i=1}^{6} \lambda_i C_i \]

\[ \frac{dC_i}{dt} = \beta_i (1+k) \frac{dQ}{dt} - \lambda_i C_i \]

\[ k_P = k(1-\beta) - \beta \]

The energy density as a function of time and space is given by

\[ E = Q \cdot N \]

The resulting reactivity is composed of contributions from an outside disturbance, the fuel temperature (Doppler) feedback, the disassembly feedback and initial conditions

\[ k = \varepsilon_{10} + \varepsilon_{20}t + \Delta k_e + \Delta k_D + \Delta k_{ex} \]
\[ \Delta k_e = \varepsilon_1 + \varepsilon_2 t \]

The reactivity density is defined by

\[ w = \frac{dk}{dV} \]

A perturbation in the reactivity density gives a perturbation in the total reactivity

\[ \delta k = \int_V \delta w \, dV \]

\[ \delta w = \alpha_D \delta T + \gamma \eta \delta T + \delta \bar{T} \, \text{grad} \, w \]

The reactivity density perturbation is composed of three perturbation terms, namely

a) Internal fuel temperature effect (Doppler effect)

\[ \alpha_D = \frac{\partial w}{\partial T} \]

b) Thermal fuel expansion effect \( (\eta = \frac{\partial w}{\partial \rho_f}) \)

neglected \( (\gamma = \frac{d \rho_f}{dT} = 0) \)

c) Mechanical fuel expansion effect (disassembly effect)

\[ \delta w_{\text{ex}} = \delta \bar{T} \, \text{grad} \, w \]

The total "Doppler" reactivity change is given by

\[ \Delta k_D = \int_V \int_T \alpha_D \, dT \]

The "Doppler" coefficient density is related to the fuel temperature according to the following weighted relation
After two time differentiations and using the impulse conservation law the total "disassembly" reactivity feedback is given by

\[ \frac{d^2 k_{ex}}{dt^2} = \frac{1}{\rho} \int p \text{div}
\n\text{grad} w \, dV \]

where grad w is obtained as a function of the energy space distribution by using first order one group perturbation theory. See appendix.

3.5 Equations of state

The equations of state relate the pressure and temperature of the fuel to the internal energy density of the fuel. The poor knowledge of some characteristics of the fuel, especially at high temperatures, leads to inaccuracies in determining these equations. At the beginning of the excursion the fuel contains a void fraction which is gradually filled by expanding fuel material in the liquid and gas phases. The pressure of the fuel gas is very small at first and is neglected before complete fuel meltdown in the infinitesimal volume considered. Then it is described by the gas pressure law of the saturated two-phase region (isobaric region). When all voids are filled or the superheated region is reached an isochoric linear relation is used. Due to the constant volume the pressure will finally rise very quickly (at high energy densities). The following adiabatic relation is used for the relation between temperature and internal energy density before core melting

\[ T = T_a + K_4 E \]

In the two-phase region the following relations are used:

\[ T = T_m + K_2 \Delta E + K_3 \Delta E^2 \]

\[ p = K_4 e^{-K_5/(\Delta E + K_6)} - K_7 - p_1 \]
For high pressures a linear relationship is used:

\[ p = K_8 (\Delta E - K_9) - p_1 \]

The cladding strength can be provided for in the calculations by assuming a suitable boundary condition for the energy density. The parameters in the equations of state have been determined by a curve fitting procedure using published curves (Ref. 2 and 3). Fig. 1 shows the fuel temperature as a function of the internal energy density after fuel-melting. Fig. 2-4 show the generated internal fuel gas pressure at different core radii versus the central internal energy density for a spherical core model.

3.6 Energy releases

The power excursion is solved by numerical integration of several time-differential equations and quasi-static relations with variables dependent on time and space. The step-by-step time calculation is stopped when the prompt reactivity has become negative enough, due to the feedbacks, to decrease the power density sufficiently. The total energy content and the energy release in the whole core are directly calculated by volume integration of the energy density. Only a part of the energy release can be used to do work. The effective nuclear energy release is defined as the total integrated energy density above a boundary level. The integration is confined to the time-variant volume inside an area where the fuel gas pressure satisfies the boundary level for the start of the mechanical core expansion disassembly. This volume represents a kind of vapour bubble, the shape of which may be calculated at any time. The effective nuclear energy release or destructive energy is

\[ E_d = Q_e \int_V^a N \, dV - \int_V^a E_{b1} \, dV \]

An upper limit of the kinetic energy or available mechanical work can be obtained by assuming an isentropic fuel gas expansion from the high pressure reached in the excursion to a low pressure state, e.g. at atmospheric pressure.
The coefficient \( k_{ad} \) (efficiency factor in converting heat energy to mechanical energy) is a function of the maximum generated fuel gas pressure and can be determined from published curves (Ref. 2) for sufficiently high pressures. Due to the normally very low pressures generated in the calculations it has not been possible to find the appropriate values of the efficiency factor. Instead upper limit values are used. For maximum pressures less than 1000 atm an upper limit ratio of 0.2 is used for the mechanical work relative to the destructive nuclear energy. The corresponding value for 10,000 atm is 0.3.

4. COMPUTER RUNS

The normal case is equivalent to a 1000 MWe fast oxide-fuel reactor without cooling, i.e. the core-meltdown is associated with a loss (voiding) of the coolant (sodium, gas or steam). The meltdown could result in an aggravation of the accident due to a core compaction which will give a secondary reactivity insertion superposing the initial primary reactivity insertion. Also the loss of the coolant may lead to a simultaneous addition of reactivity. This is particularly the case for the sodium cooled fast reactor due to the sodium superheating before sudden boiling. Ramp reactivity rates from \( 0.05 \text{ s}^{-1} \) to \( 0.45 \text{ s}^{-1} \) have been used in the calculations. The initial conditions are full nominal power (2500 MWth) and a fuel temperature in the core centre equal to the fuel melting temperature (3070 K). In order to demonstrate the general shape of the reactivity, nuclear power and disassembly energy yield excursions for a ramp rate of \( 0.35 \text{ s}^{-1} \) (100 \( \beta \)/s), the curves in fig. 5, 6 and 7 are shown. There is an oscillating behaviour of the power and reactivity due to the interaction between the Doppler reactivity and the inserted ramp reactivity. The average power level is determined by the condition of zero average prompt reactivity and is, in general, given by the ratio between the inserted ramp rate and the energy reactivity coefficient (Doppler feedback). After 15 ms the disassembly starts which finally will lower the reactivity and thus terminate the power excursion. The disassembly effect on the reactivity is delayed, how-
ever, due to the low initial pressures generated in the saturated region, the small part of the core initially taking part in the disassembly and the inertia of the core masses involved. The decreasing Doppler reactivity coefficient at higher temperatures, therefore, counteracts the disassembly effect at the beginning of the excursion and causes an increase of the power bursts as shown in fig. 6. The disassembly part of the excursion lasts about 10 ms as shown in fig. 7. In order to provide a check of the calculations the computations were repeated for initial conditions of the reactivity and power level density corresponding to the moment of the start of the disassembly phase of the excursion. These initial conditions were given as outputs from earlier computer runs starting from the initial conditions at the start of the reactivity insertion (either primary reactivity insertion with intact core or including a secondary reactivity insertion due to a partial or total core collapse). The good agreement between the different calculations, as shown in fig. 7, proves that the code gives an accurate description of the disassembly phase once the two mentioned initial conditions are established, which could and should be done by a separate meltdown code giving a dynamic treatment of the impact of the meltdown process on the power and reactivity transients. Also the initiation of the accident and the transients leading to meltdown should be studied by the normal reactor dynamic codes in order to establish initial conditions for a meltdown code. In this report all efforts have been concentrated on the disassembly phase of the excursions. The total nuclear energies released since that start of the excursion and the disassembly ($E_T$ and $E_{TE1}$ respectively) are linear functions of the central internal energy density ($Q$) as shown in fig. 8. Only a fraction of the core takes part in the disassembly and the nuclear energy released in that part ($E_{TE2}$) is also shown. Subtracting the energy released before disassembly for all parts of the core the rest is equivalent to the effective disassembly nuclear energy release ($E_{TE3} = E_d$) available for conversion into a mechanical "damage" energy release. This destructive energy is considered as the final output for the EEM model and the parametric dependence of the destructive energy has been studied using a normalized reactor core and normalized initial conditions at the start of the disas-
assembly. The oscillations normally encountered in the study of parametric energy yield variations have therefore been largely eliminated and, by choosing conservative values for the normalized initial conditions, the energy releases calculated are close to the upper envelopes of the parametric dependences. The oscillations are partly a result of the mentioned oscillating behaviour of the power and reactivity during the predisassembly phase of the excursion and partly a result of the large dependence of the energy releases on the actual initial conditions at the start of the disassembly, e.g. increasing the initial conditions of reactivity and power to peak values will result in a destructive energy release of 61% of the normalized value for a studied parameter set.

The dependence of the normalized destructive nuclear energy release on the ramp reactivity is shown in fig. 9 (the normalized value 0.35 s⁻¹ corresponds to 100 $\beta$/s) and is approximately linear. The dependence on the central specific Doppler reactivity coefficient at meltdown is shown in fig. 10 (the normalized value $0.82 \times 10^{-12} \text{ g}^{-1} \text{ K}^{-1}$ corresponds to a total uniform central Doppler reactivity constant of 0.031 at meltdown or a total Doppler reactivity coefficient of $0.42 \times 10^{-5} \text{ K}^{-1}$ at central meltdown) and is roughly inversely linear. The power density space distribution is assumed time-independent (the normalized distribution for a spherical geometry is shown in fig. 11 and corresponds to a power form factor of 1.9). The dependence of the normalized destructive nuclear energy release on the power form factor is shown in fig. 12. The energy release increases rapidly with decreasing form factor ($F_p$):

$$F_p \quad 1.9 \quad 1.5 \quad 1.1$$

$$E_d \quad 0.21 \quad 0.38 \quad 0.93 \quad J \times 10^{10}$$

The dependence of the normalized destructive nuclear energy release on the core volume is shown in fig. 13. The two curves correspond to different initial conditions. The oscillating curve implies a steady-state condition when the normalized ramp reactivity rate ($0.35 \text{ s}^{-1}$) is imposed on the system, whereas the "smooth" curve implies the normalized set of initial conditions at the start of disassembly as mentioned earlier. Fig. 13 clearly shows the importance of establishing correct initial conditions for a given accident in order to make even a rough
estimation of the destructive energy release for a given reactor. Due to the uncertainties of the various reactor parameters, the thermodynamic constants and the accidental mechanisms, it is therefore obviously satisfactory to limit the calculation to the estimation of upper limit values using a simple mathematical model such as the one used in this report. It is also obviously unrealistic at the present time to perform detailed studies of variations in core configurations, cooling and fuel media and zoning. Such calculations would require an extension of the mathematical model to include the detailed melting behaviour and the development of a fictitious perturbation model without phase-sensitive oscillations. The studies performed give the orders of magnitude of the mechanical shocks the containment will be subjected to and also give some suggestions as to how to reduce these shocks.

5. CONCLUSIONS

The importance of the available Doppler feedback for minimizing the consequences of a prompt reactivity accident must be stressed. The available Doppler feedback \( \Delta k_{Dop} \) is the absorbed reactivity when the core temperature increases very quickly from the normal operating conditions to a point where serious damage occurs or conservatively to the point of hot spot melting. The allowable rapid reactivity insertion magnitude is less than \( 1 + \Delta k_{Dop} \). For slow reactivity insertions the excursion is limited by the normal feedback effects (before core melting). In that case the Doppler feedback will give an upper limit of the allowable insertion rate. An internally safe fast reactor is thus a reactor with a large available Doppler feedback which is achieved by using a low maximum fuel operative temperature, a large power form factor and a large negative Doppler reactivity constant. The core compaction accident is then less probable and the energy releases following a less probable core disassembly are considerably reduced.

It is also important to note the effect of a large power form factor in reducing the destructive energy release following a core disassembly. It is suggested to introduce small local zones, preferably in the core centre, with higher power densities (or lower melting temperatures), i.e. one such zone in the reactor centre would act like a fuse shutting
down the reactor in the case of a reactivity accident before a serious compaction and explosive disassembly excursion could occur. The shutdown effect is accomplished firstly by the reactivity reduction associated with the displacement of fuel towards a less active location (negative displacement reactivity feedback) and secondly by the development of a very small disassembly nuclear excursion, should the first effect not be sufficient.
APPENDIX

General expressions for the explosive expansion reactivity effect

Integrating over the whole core space the total expansion reactivity perturbation is given by

\[ \delta k_{\text{ex}} = \int_V \delta \mathbf{r} \cdot \nabla \psi \, dV \]

After one time differentiation

\[ \frac{d}{dt} \delta k_{\text{ex}} = \int_V \mathbf{v} \cdot \nabla \psi \, dV \]

\[ \mathbf{v} = \frac{d\mathbf{r}}{dt} \] is the velocity of the core material during the disassembly. Neglecting friction effects and the gravity the hydraulic impulse conservation law reads

\[ \int_0^t \frac{dv}{dt} + \nabla p = 0 \]

Supposing a time-independent grad \( \psi \) another time differentiation will therefore give

\[ \frac{d^2 k_{\text{ex}}}{dt^2} = \int_V \frac{1}{\rho} \nabla p \cdot \nabla \psi \, dV \]

The following vector formulas can be used:

\[ \nabla \cdot (\rho \nabla \psi) \, dV = \int_\partial V \hat{n} \cdot \nabla \psi \, dS \]

\( \hat{n} \) is a unity space vector perpendicular to the area fragment \( dS \) and directed outwards from the volume \( V' \).
For a space-independent core mass density ($\rho$) the following equation is derived:

$$\frac{d^2 k_{\text{ex}}}{dt^2} = \frac{1}{\rho} \int V \rho \text{div} \text{grad} \, w \, dV - \frac{1}{\rho} \sum_i S_i \hat{n} \rho \text{grad} \, w \, dS$$

The expansion volume of the core is denoted by $V_E$ and is surrounded by an area $S_E$. The pressure immediately inside the area is denoted by $p_{\text{in}}$ and outside the area by $p_{\text{out}}$. Therefore (one-zone core):

$$\frac{d^2 k_{\text{ex}}}{dt^2} = \frac{1}{\rho} \int V \rho \text{div} \text{grad} \, w \, dV + \frac{1}{\rho} \sum S_E \left[ p_{\text{out}} (\text{grad} \, w)_{\text{out}} - p_{\text{in}} (\text{grad} \, w)_{\text{in}} \right] \cdot \hat{n}_E \, dS$$

Neglecting the pressure gradient change at the surface one has $p_{\text{out}} = p_{\text{in}}$ and $(\text{grad} \, w)_{\text{out}} = (\text{grad} \, w)_{\text{in}}$. Using a first order one group perturbation theory the following equation is derived (one-zone core):

$$\text{grad} \, w = AD \left[ \text{grad} \left( \text{grad} \, N \right)^2 - 2 \text{div} \text{grad} \, N \, \text{grad} \, N \right]$$

The diffusion constant ($D$) is assumed space-independent

$$A^{-1} = F^{-1} \nu \Sigma_f \int V N^2 \, dV$$

$F$ = core fission fraction
$
\nu$ = neutron multiplication constant (space-independent)
$
\Sigma_f$ = macroscopic fission cross section (space-independent)

The equations above are combined:

$$\frac{d^2 k_{\text{ex}}}{dt^2} = \frac{A \Sigma_f}{\rho} \int V \rho \text{div} \left[ \text{grad} \left( \text{grad} \, N \right)^2 - 2 \text{div} \text{grad} \, N \, \text{grad} \, N \right] \cdot dV$$

In spherical geometry:
\[
\frac{d^2k_{ex}}{dt^2} = \frac{4\pi}{\rho} \int_0^a pr^2 \left[ \frac{\partial}{\partial r} (\text{grad } w)_r + \frac{\partial}{\partial r} (\text{grad } w) \right] dr
\]

\[
\text{grad } w = -4AD \frac{r}{r} \left( \frac{\partial N}{\partial r} \right)^2
\]

\[
A^{-1} = 4\pi F^{-1} \int \left[ r^2 N^2 \right] dr
\]

\[
\frac{d^2k_{ex}}{dt^2} = -16\pi \frac{AD}{\rho} \int_0^a \frac{\partial N}{\partial r} \left( \frac{\partial N}{\partial r} + 2r \frac{\partial^2 N}{\partial r^2} \right) dr
\]

\[a = \text{core radius}\]
\[a_E = \text{time-variant core "disassembly" radius}\]

The one group neutron diffusion equation gives the following static space variation of the neutron flux in spherical symmetry:

\[N = \frac{1}{Br} \sin (Br) \quad r \leq a\]

\[B = \frac{\pi}{a+d}\]

B = buckling
\[d = \text{extrapolation distance.}\]

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Fig. 2

Generated fuel gas pressure vs. central energy density at various core radii (spherical geometry)
Fast power reactor - oxide fuel
Large scale-factors
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Generated fuel gas pressure vs. central energy density at various core radii (spherical geometry)
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General fuel gas pressure vs. central energy density at various core radii (spherical geometry)
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Fig. 5
Excess prompt reactivity transient for $0.35 \text{ s}^{-1}$ ramp reactivity insertion

$[10^5] k_p$

$t$ (ms)

-500

-400

-300

-200

-100

0

10

20

30
Nuclear power transient for 0.35 s⁻¹ ramp reactivity insertion

\( P_o = \text{central power density} \) - \( \frac{P_o}{P_{ao}} \) = normalized value (= 1 at the excursion start) - \( P_T \) = total power.
Effective nuclear energy release transient (during core disassembly) for 0.35 s$^{-1}$ ramp reactivity insertion.

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Total nuclear energy releases vs. central energy density

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