Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

REGULATORY GUIDE

Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants

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Canada

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Regulatory documents support the Canadian Nuclear Safety Commission (CNSC) regulatory framework. By expanding on expectations set out in general terms in the *Nuclear Safety and Control Act* and associated regulations, regulatory documents provide one of the core management tools upon which the CNSC relies to fulfill its legislated obligations.

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Regulatory Notice (N): The regulatory notice notifies licensees and other stakeholders about significant matters that warrant timely action.

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TRIP PARAMETER ACCEPTANCE CRITERIA FOR THE SAFETY ANALYSIS OF CANDU NUCLEAR POWER PLANTS

1.0 PURPOSE

The purpose of this Regulatory Guide is, in the interest of achieving the objectives of the *Nuclear Safety and Control Act* (NSCA), to provide guidance to licensees who operate CANDU nuclear power plants regarding reactor trip parameters that will preclude direct or consequential failures of reactor fuel or reactor pressure tubes.

2.0 SCOPE

This Regulatory Guide outlines the criteria that the selected reactor trip parameters for the plant are expected to meet, under all postulated design basis accidents other than the following:

1. Large loss of coolant accidents (LLOCA);
2. Very slow loss of reactivity control (VSLORC) accidents;
3. Fast loss of reactivity control (FLORC) accidents;
4. Fuelling machine accidents;
5. Single channel accidents[1]; and
6. Accidents in the spent fuel bay.

The acceptance criteria described in this guide are intended to be used in the safety analysis for fuel bundles that have experimental post-dryout heat transfer data with a relatively low post-dryout fuel sheath temperature. For the postulated plant accidents, the experimental data used in the analysis for dryout and post-dryout heat transfer should cover a range of conditions and phenomena that reasonably and credibly account for in-reactor conditions.

3.0 RELEVANT LEGISLATION

Provisions of the NSCA and the regulations made under the NSCA that are most pertinent to this Regulatory Guide are as follows:

1. Paragraph 3(a) of the NSCA states that the purpose of the NSCA is to provide for “the limitation, to a reasonable level and in a manner that is consistent with Canada’s international obligations, of the risks to national security, the health and safety of persons and the environment that are associated with the development, production and use of nuclear energy”;

[1] The current, stylized, limit-of-operating-envelope (LOE) analysis assumes a priori fuel failure.

2. Subsection 24(4) of the NSCA stipulates that “No licence may be issued, renewed, amended or replaced unless, in the opinion of the Commission, the applicant
   a) is qualified to carry on the activity that the licence will authorize the licensee to carry on; and
   b) will in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed”;
3. Subsection 24(5) of the NSCA, provides that a licence issued by the Commission may contain any term or condition that the Commission considers necessary for the purposes of the NSCA;
4. Paragraph 12(1)(*c*) of the *General Nuclear Safety and Control Regulations* requires that every licensee “take all reasonable precautions to protect the environment and the health and safety of persons and to maintain security”; and
5. Paragraph 12(1)(*f*) of the *General Nuclear Safety and Control Regulations* requires that every licensee “take all reasonable precautions to control the release of radioactive nuclear substances or hazardous substances within the site of the licensed activity and into the environment as a result of the licensed activity.”

4.0 BACKGROUND

The performance requirements for the reactor shutdown system(s) for all design basis accidents are such that the fuel integrity[2] and the primary heat transport system integrity should not be jeopardized. The avoidance of fuel sheath dryout as a surrogate to fuel and consequential pressure tube failure is an acceptable approach.

4.1 Potential Consequences

Fuel sheath dryout has potential consequences that include failure of the fuel, failure of the fuel and pressure tube, or failure of the pressure tube. These outcomes may be prevented by limiting the post-dryout operation. Under post-dryout conditions, fuel sheath heatup rate and the extent of fuel deformation may have significant safety implications. The safety implications would be benign if the rate and the extent of fuel deformation are small. Conversely, if the heatup rate and extent of fuel deformation are large, the pressure tube could fail due to local heating of the pressure tube. Local heating may be due to either fuel failures or fuel element(s) making contact with the pressure tube.

[2] This requirement does not apply to LLOCA and single channel design basis events. Fuel failures are expected to occur for these events.

4.2 Fuel Sheath Dryout

Available information (see References, page 8, #3) indicates that fuel sheath dryout over a short period, combined with a gradual heatup, is unlikely to cause fuel sheath failures. If the maximum fuel sheath temperature is below 600^{o}C, and the duration of post-dryout operation is less than 60 seconds, it is accepted that the fuel deformation will be small and that the fuel elements will not contact the pressure tube to cause a failure of the pressure tube.

4.3 Safety Analysis Methodology

The choice of the safety analysis methodology to demonstrate plant safety rests with the licensee. The analysis may be performed at the limit of operating envelope (LOE) or a best estimate and uncertainty (BEAU) methodology may be used. If a BEAU-type of analysis methodology is used, the acceptance criteria should be met at a certain level of probability and confidence limit commensurate with the risk posed by the postulated event.

5.0 TRIP PARAMETER ACCEPTANCE CRITERIA

In nuclear power plants, there are systems that measure and monitor the values of important plant parameters (such as pressure, temperature, neutron flux, etc.). These parameters, the values of which are measured by the trip system and used to shut down the reactor, are called the trip parameters.

Under plant upset or accident conditions, the values of some of these parameters would exceed prescribed limits. The reactor trip system continuously measures the values of these important parameters and initiates a plant shutdown action when the measured values exceed prescribed limits.

A licensee may adopt the following trip parameter acceptance criteria in the licensee’s safety analysis to demonstrate that direct or consequential failures of reactor fuel or reactor pressure tube failures due to any fuel failures are precluded.

The trip parameter acceptance criteria include the following:

1. The primary trip parameter predefined limit on each shutdown system should be selected so as to prevent the onset of intermittent fuel sheath dryout; and
2. The backup trip parameters predefined limit on each shutdown system should be selected so as to prevent:
    a) fuel sheath temperature from exceeding 600^{o}C, and
    b) the duration of post-dryout operation from exceeding 60 seconds.

5.1 Revisions to the Acceptance Criteria

The trip parameter acceptance criteria in section 5.0 may be revised by the CNSC if the fuel bundle experimental database is extended to a significantly higher fuel temperature beyond the current post-dryout heat transfer regime. The experimental data should clearly demonstrate[3] that the nominal fuel geometry and fuel sheath integrity are maintained under conditions comparable to reactor accident conditions while the pressure tube integrity is not jeopardized under these conditions.

6.0 CONDITIONS FOR POSTULATED REACTOR ACCIDENTS

The trip parameter acceptance criteria described in section 5.0 above will be used by the CNSC staff to assess the acceptability of the trip parameters for all postulated design basis reactor accidents[4] (except those listed in section 2.0) for which the following two conditions are satisfied:

1. $\mathbf{0 < t_{DO} < t_{FCM}}$; and
2. $\mathbf{(t_{FCM} - t_{DO}) \geq (x + y)}$ **seconds**

where, $\mathbf{t_{DO}}$ = Time to fuel sheath dryout, in seconds

$\mathbf{t_{FCM}}$ = Time to fuel centerline melting, in seconds

$\mathbf{x}$ = 60 seconds

$\mathbf{y}$ = requirement on the shutdown system shut off rod insertion rate[5], in seconds.

[3] The experimental data should clearly demonstrate a high level of confidence while taking into account all of the uncertainties. These include deviations from the design values for the fuel sheath, pellet, and bundle (bundle deformation).

[4] These are events with a frequency of occurrence of 10^{-2}/year and higher, with some exceptions.

[5] Traditionally, the trip parameter success criterion for fast loss of reactivity and large loss-of-coolant accident is t_{FCM} - 1.5seconds. For LLOCA and FLORC accidents, it is assumed that a fast shutdown system action for full insertion of shutoff rods (SORs) in 1.5 seconds just prior to the time of fuel centreline melting (FCM) will prevent FCM. This represents the integrated energy deposition on fuel between time zero (start of the initiating event) until the time of full insertion of the SORs into the core. This integrated energy would be less than the energy corresponding to FCM temperature. Noting that this requirement is derived from the rate of energy deposition into the fuel (accident specific), the most recent acceptable value must be substituted for “y”.

GLOSSARY

best estimate and uncertainty (BEAU) methodology
A term for an analysis performed using models that attempt to realistically describe the physical processes occurring in a nuclear reactor with the uncertainties quantified and accounted for in the prediction of the key plant parameters. A key concept in any best estimate methodology is the capability to realistically predict plant behaviour and important parameters.

In order to meet the derived safety acceptance criteria to a prescribed probability and level of confidence, all the uncertainties due to modeling, coding, correlations, and plant initial and boundary conditions should be explicitly accounted for, and propagated throughout the plant transient simulations.

consequences
Within the nuclear industry, “consequences” suggests undesirable results or outcomes; e.g., the release of the radioactive fission products into the reactor building and/or escape of the radioactive products to the environment.

design basis accident
Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

fast loss of reactivity control (FLORC)
In nuclear reactors, there are devices to measure, control, and maintain reactivity by inserting or removing reactivity devices. One of the hypothetical accidents that a nuclear reactor should be protected against (by design) is a loss of this control function. Only loss of reactivity control accidents that result in reactor power increase pose a safety concern.

fuel deformation
Change in the geometry of the fuel bundle brought about by deformation of one or more elements in the bundle or deformation of the bundle as a whole.

fuel sheath dryout
Under normal operating conditions, the fuel elements are cooled by a liquid coolant flowing over the fuel. Under some accident conditions, the liquid coolant may boil off and form a vapour blanket over the fuel sheath. This is termed as fuel sheath dryout.

Fuel sheath dryout causes the temperature of the fuel (and the sheath) to increase and the formation of more vapour, which, in turn, further increases the fuel temperature.

heatup rate
This term is used in reference to the source of energy (i.e., the fuel) that provides energy (via nuclear reaction) in a controlled manner whereby a constant fuel temperature is maintained under normal operating conditions. However, under accident conditions, the production of energy exceeds the removal of energy (from the fuel) giving rise to an increase in fuel temperature. The rate at which the energy increases is the heatup rate.

large loss of coolant accidents (LLOCA)
This is part of a category of accident analyses that deals with reactor safety due to a loss of coolant from the primary heat transport system. A large loss of coolant accident results from a large size break in the primary heat transport system.

limit of operating envelope (LOE)
This term is used for a deterministic safety analysis that assumes, prior to the postulated accident, that the plant was operating with some of the important plant operating parameters being at their safety limits, while some of the models used to describe the event may be conservative. The qualifier "deterministic" that is used as a prefix to the word analysis, means that the analysis is done using prescribed and what is perceived to be conservative assumptions to account for uncertainties in the models, codes, correlations, and initial and boundary conditions of the plant.

LOE does not necessarily mean an impossible plant operating state. However, depending on the number and nature of the conservative assumptions made in the analysis of the event, it may become a highly improbable, if not a physically impossible, event.

nuclear power plant
Any fission reactor installation that has been constructed to generate electricity on a commercial scale. A nuclear power plant is a Class IA nuclear facility, as defined in the *Class I Nuclear Facilities Regulations*.

post-dryout
Under plant upset or abnormal plant operating conditions, the fuel sheath may dryout (see fuel sheath dryout). If operator action(s) and/or the reactor regulating system are ineffective, the automatic shutdown system may shut the reactor down. Starting from the time of the first incipient of fuel sheath dryout until the time of reactor shutdown, continued high power operation is termed as post-dryout operation.

primary and backup (secondary) trip parameter
The designation of a trip parameter as the primary or backup trip parameter is established by means of safety analyses. By definition, the trip parameter that is predicted earliest in time on each shutdown system is the primary trip parameter. Similarly, the trip parameter that is predicted to come after the primary trip parameter (on each shutdown system) is called the backup trip parameter.

SORs (shutoff rods)
In nuclear power plants, under any plant upset or accident condition, there are automatic detection systems equipped with neutron absorbing solid metallic rods that are inserted into the reactor core to stop (shut off) the nuclear reaction. These metallic rods are referred to as shutoff rods.

very slow loss of reactivity control (VSLORC)
Industry abbreviations for VSLORC are Neutron Over Power (NOP) and Regional Over Power (ROP). NOP/ROP type of accidents are a small subset of larger loss of reactivity control accidents that have always been analyzed separately using a comparable best estimate methodology with a statistical approach to the requirement of prevention of fuel sheath dryout. The reactor trip parameter for this type of postulated event is called the NOP (or the ROP) trip parameter. This is a highly stylized design basis accident and has been traditionally accepted to be covered by a single trip parameter only.

REFERENCES

1. *Nuclear Safety and Control Act,* S.C. 1997, c. 9.

2. *General Nuclear Safety and Control Regulations*, SOR/2000-202.

3. Leung, A., Segel, A.W.L. and Merlo, E.E., Atlantis Engineering. *Assessment of Bundle Deformation under Post-Dryout Condition*. Ottawa: Atomic Energy Control Board, 1994. Report prepared for the AECB (research project no. 2.304.1).

4. *Class I Nuclear Facilities Regulations,* SOR/2000-204.