Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

DRAFT
REGULATORY
STANDARD

Safety Analysis for Non-Power Reactors

S–308

Issued for Public Consultation
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Canada

TYPES OF REGULATORY DOCUMENTS

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.

The regulatory documents most commonly published by the CNSC are *regulatory policies*, *regulatory standards*, and *regulatory guides*. At the highest level, regulatory policies provide the direction for regulatory standards and guides, which serve as the policy "instruments." A fourth type of regulatory document, the *regulatory notice*, is issued when warranted. Because the information in a *regulatory notice* must be conveyed with relative urgency, the development process is faster than that applied to the other documents.

Regulatory Policy (P): The regulatory policy describes the philosophy, principles or fundamental factors on which the regulatory activities associated with a particular topic or area of concern are based. It describes why a regulatory activity is warranted, and therefore promotes consistency in the interpretation of regulatory requirements.

Regulatory Standard (S): The regulatory standard clarifies CNSC expectations of what the licensee should do, and becomes a legal requirement when it is referenced in a licence or other legally enforceable instrument. The regulatory standard provides detailed explanation of the outcomes the CNSC expects the licensee to achieve.

Regulatory Guide (G): The regulatory guide informs licensees about how they can meet CNSC expectations and requirements. It provides licensees with a recommended approach for meeting particular aspects of the requirements and expectations associated with their respective licensed activities.

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

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SAFETY ANALYSIS FOR NON-POWER REACTORS

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About this Document

This draft regulatory standard sets out the requirements related to the safety analysis and the conducting of safety analysis, including the selection of events to be analysed, acceptance criteria, safety analysis methods, and safety analysis documentation and review of fission reactors that are not nuclear power plants.

Comments

The CNSC invites interested persons to assist in the further development of this draft regulatory document by commenting in writing on its content and potential usefulness. Please respond by November 15, 2006. Direct your comments to the postal or e-mail address provided below, referencing file 1-8-8-308.

The CNSC will take the comments received on this draft into account when developing it further. Any comments submitted including names and affiliations, may be made public.

Document availability

This document can be viewed on the CNSC Internet web site at (www.nuclearsafety.gc.ca). To order a printed copy in English or French, please contact:

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SAFETY ANALYSIS FOR NON-POWER REACTORS

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SAFETY ANALYSIS FOR NON-POWER REACTORS

1.0 PURPOSE

The purpose of this regulatory standard, when incorporated in a licence or other legally enforceable instrument, is to help assure, during the preparation of a site for a fission reactor that is not a nuclear power plant, or during the construction, operation or decommissioning of a fission reactor that is not a nuclear power plant, that adequate safety analyses are completed by, or on behalf of, the licensee in accordance with defined regulatory requirements.

2.0 SCOPE

This regulatory standard sets out the requirements related to the safety analysis and the conducting of safety analysis, including the selection of events to be analysed, acceptance criteria, safety analysis methods, and safety analysis documentation and review.

This standard applies to reactors whose primary purpose is to carry out research and tests or to produce isotopes, including reactor types such as NRU, MNR, ZED-2, SLOWPOKE, MAPLE and École Polytechnique’s subcritical assembly.

3.0 RELEVANT LEGISLATION

The relevance of the *Nuclear Safety and Control Act* (NSCA) and the regulations made under the NSCA to this Standard is as follows:

1. Subsection 24(4) of the NSCA provides that the Commission may only issue, renew or amend licences if the licensee or the applicant “(a) is qualified to carry on the activity that the licence authorize the licensee to carry on, and (b) will, in carrying out 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”;
2. Subsection 24(5) of the NSCA authorizes the Commission to include in a licence any term or condition that the Commission considers necessary for the purposes of the NSCA;
3. Paragraph 3(1)(*i*) of the *General Nuclear Safety and Control Regulations* stipulates that an application for a licence shall contain, in addition to other information, “a description and the results of any test, analysis or calculation performed to substantiate the information included in the application”;
4. Paragraph 5(*f*) of the *Class I Nuclear Facilities Regulations* provides that an application for a licence to construct a Class I nuclear facility shall contain, in addition to other information, information on “a preliminary safety analysis report demonstrating the adequacy of the design of the nuclear facility”;

5. Paragraph 5(*i*) of the *Class I Nuclear Facilities Regulations* provides that an application for a licence to construct a Class I nuclear facility shall contain, in addition to other requirements, information on “the effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility...”;
6. Paragraph 6(*c*) of the *Class I Nuclear Facilities Regulations* provides that an application for a licence to operate a Class I nuclear facility shall contain, in addition to other requirements, information on “a final safety analysis report demonstrating the adequacy of the design of the nuclear facility”;
7. Paragraph 6(*h*) of the *Class I Nuclear Facilities Regulations* that stipulates that an application for a licence to operate a Class I nuclear facility shall contain, in addition to other requirements, information on “the effects on the environment and the health and safety of persons that may result from the operation and decommissioning of the nuclear facility...”; and
8. Paragraph 7(*f*) of the *Class I Nuclear Facilities Regulations* that provides that an application for a licence to decommission a Class I nuclear facility shall contain, in addition to other requirements, information on “the effects on the environment and the health and safety of persons that may result from the decommissioning of the nuclear facility....”

4.0 BACKGROUND

Non-power reactors licensed by the CNSC employ a variety of designs and a wide range of operating power levels. Use of these reactors also covers a wide range of purposes, from simple irradiations of target materials for neutron activation analysis, to complex experiments involving irradiation testing of fuels and materials.

For licensing of a specific facility design, licensees may use the graded approach introduced in the International Atomic Energy Agency (IAEA) *Safety Requirements of Research Reactors* (DS 272) when the safety analysis is conducted. The graded approach is a risk-informed approach that calls for the safety analysis requirements to be commensurate with the risk of the facility. A licensee may use the graded approach to determine the scope, extent and detail to be followed for the safety analysis. Such an approach facilitates the regulatory review process by reducing any unnecessary burden to the licensee.

Safety analysis typically involves deterministic and probabilistic analyses in support of the siting, design, commissioning, operation, refurbishment or decommissioning of a nuclear facility. This standard focuses on the deterministic safety analysis (hereafter called safety analysis) used in the assessment of event consequences.

5.0 SAFETY ANALYSIS OBJECTIVES

The objectives of the safety analysis are to:

1. Confirm that the design of a nuclear facility meets design and safety requirements;
2. Derive or confirm operational limits and conditions which are consistent with the design and safety requirements for the facility;
3. Assist in establishing and validating accident management procedures and guidelines;
4. Assist in demonstrating that safety goals, which may be established to limit the safety risks posed by the nuclear facility, are met; and
5. Confirm that changes in the facility have no adverse impact on safety.

This standard identifies high level requirements for conducting and presenting a safety analysis, based on the best national and international practices.

The results of a safety analysis are used to specify operational limits and conditions (OLCs), and to provide input to the commissioning program, operating procedures, periodic inspection and testing, maintenance, emergency operating procedures and accident management plan for the facility.

6.0 SAFETY ANALYSIS REQUIREMENTS

6.1 Responsibility

The licensee is responsible for ensuring that the safety analysis meets all regulatory requirements. The licensee shall:

1. Maintain adequate capability to perform or procure safety analysis;
2. Establish a formal process to assess and update a safety analysis, accounting for the impact of design modifications, operational experience, research findings and identified safety issues; and
3. Establish and apply a documented quality assurance (QA) process for conducting a safety analysis.

6.2 Events to be Analysed

6.2.1 Identifying Events

The licensee shall identify, using a comprehensive systematic process, initiating events (including criticality events), event sequences and event combinations ("events" hereafter in this document) that can potentially challenge the safety or control functions of the facility. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and systematic review of the design.

The identification of events shall account for all operating states, configurations and uses of the facility. The interaction between the reactor and the experimental devices, including administrative procedures, controls and provisions related to the experimental devices, shall be accounted for. Materials that are allowed in experiments that are performed in or near the reactor core, together with materials that may only be used under additional safety conditions, shall be identified.

The list of identified events shall be reviewed for completeness during the design and analysis process. Any subsequent design changes or experiment designs will be reviewed and the list of identified events shall be modified as necessary.

In addition to events that challenge the safety or control functions of the reactor, the safety analysis shall be performed for the normal operation of the facility.

6.2.2 Scope of Events to be Analysed

The list of events to be developed for the safety analysis shall include:

1. Component and system failures or malfunctions;
2. Operator errors; and
3. Common cause internally and externally initiated events.

The main safety function that protects the reactor core from failure of experimental devices or samples shall be identified and the initiating events that can challenge this safety function, together with any event sequences, shall be established.

A cut-off frequency shall be selected such that the events with a frequency of occurrence less than the cut-off limit would provide only a negligible contribution to the risk. The elimination of such events from the analysis scope shall be justified and the reasons for eliminating them documented.

6.2.3 Classification of Events

The identified events shall be classified, based on the results of probabilistic studies and engineering judgement, into the following three classes of events:

1. Anticipated Operational Occurrences (AOO) include all events with frequencies of occurrence equal to or greater than 10^{-2} per reactor year.
2. Design Basis Accidents (DBA) include all events with frequencies of occurrence equal to or greater than 10^{-6} per reactor year but less than 10^{-2} per reactor year. This class of events also includes any events that are used as a design basis for a safety system, regardless of whether the estimated frequencies are less than 10^{-6} per reactor year; and
3. Beyond Design Basis Accidents (BDBA) include events with frequencies of occurrence less than 10^{-6} per reactor year.

Events with a frequency on the border between two classes of events, or with substantial uncertainty over the predicted event frequency, shall be classified into a higher frequency class.

Credible common cause events shall also be classified within the AOO, DBA and BDBA classes.

6.3 Acceptance Criteria

6.3.1 Normal Operation

Analysis for normal operation of the facility shall demonstrate the following:

1. Facility parameter values do not exceed OLCs;
2. Radiological doses to workers and members of the public are within the limits prescribed in the *Radiation Protection Regulations*; and
3. Releases of radioactive materials into the environment are within the allowable limits.

6.3.2 Anticipated Operational Occurrences and Design Basis Accidents

Analysis for AOO and DBA shall demonstrate the following:

1. As a minimum, the applicable safety requirements specified in Table A.1 in Appendix A are met; and
2. For events where the initiating event postulates failure of fuel or the fuel site, radiological doses to workers and members of the public are within regulatory limits.

6.3.3 Beyond Design Basis Accidents

Analysis for BDBA shall demonstrate the following:

1. The facility as designed is capable of meeting the established safety goals; and
2. Accident management program and design provisions, in place to handle the accident management needs, are effective.

6.3.4 Application of Safety Requirements for AOO and DBA

The baseline safety requirements for AOO and DBA are identified, qualitatively, in Table A.1 in Appendix A. Additional requirements may be applied to reflect events resulting from unique facility design, or experiments.

To demonstrate that the qualitative safety requirements are met, acceptance criteria specific to each analysed event shall be identified prior to performing the analysis. Such acceptance criteria shall be justified and supported by appropriate evidence.

The results of a safety analysis shall meet appropriate acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis.

The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of acceptance criteria (i.e., the limiting event in an event category).

6.4 Safety Analysis Methods and Assumptions

6.4.1 General

The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria.

To achieve the appropriate level of confidence, the safety analysis shall:

1. Be performed in accordance with an approved QA process;
2. Be performed by qualified analysts;
3. Apply a systematic analysis method;
4. Use verified and validated models and computer codes;
5. Use justified assumptions;
6. Account for uncertainties in the safety analysis models and inputs;
7. Build in a degree of conservatism commensurate with the level of knowledge related to simulating the event; and
8. Be subjected to an independent or peer review.

6.4.2 Analysis Method

The analysis method shall include:

1. Identifying the scenarios to be analysed as required to attain the analysis objectives, including sensitivity cases;
2. Identifying the applicable acceptance criteria and limits;
3. Collecting the information that describes the analysed facility and its permissible operating modes;
4. Defining the assumptions regarding the operating state, the availability and performance of the facility systems, and actions of the operators;
5. Identifying the important phenomena of the analysed accident transients;
6. Selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications;
7. Identifying significant uncertainties associated with system performance, operational measurements, and facility and accident modelling;
8. Preparing input data for the analysis;
9. Conducting calculations, including sensitivity cases, to predict the event transient, starting from the initial steady state up to the pre-defined end-state;
10. Verifying calculation results for physical and logical consistency; and
11. Processing and documenting results of the calculations to demonstrate conformance with the acceptance criteria.

6.4.3 Analysis Assumptions

Safety analysis shall be based on complete and accurate facility design and operational information and supported by experimental data. Assumptions made to simplify the analysis, as well as assumptions concerning the availability and performance of the systems and operators shall be identified and justified.

The safety analysis shall:

1. Incorporate sufficient margins in the analysis assumptions to offset uncertainties associated with system performance, operational measurements, and facility and accident modelling;
2. Apply the single-failure criterion to all safety systems and their support systems;
3. Use minimum allowable performance for safety systems and their support systems;
4. Account for consequential failures that may occur as a result of the initiating event;
5. Credit actions of systems only where the systems are qualified for the accident conditions or when their actions may have a detrimental effect on the consequences of the analysed accident;

6. Consider effects of aging of components, systems and structures;
7. Account for the possibility of the equipment being taken out of service for maintenance; and
8. Credit operator actions only when there are:
    a) Unambiguous indications for such actions,
    b) Adequate procedures and operator training for such actions,
    c) Sufficient time to perform the credited actions, and
    d) Environmental conditions that do not prohibit such actions.

6.4.4 Computer Codes

Computer codes used in the safety analysis shall be developed, validated and used in accordance with a quality assurance program that meets the Canadian Standards Association CSA standard N286.7. The CNSC regulatory document, G-149 provides guidance on computer code requirements.

6.5 Safety Analysis Documentation

The safety analysis documentation shall be comprehensive, able to withstand close scrutiny, unambiguous and self-consistent. The documentation shall be sufficiently detailed so that a qualified specialist can understand the documentation without recourse to the originator. Consistent nomenclature shall be used throughout the documentation. The document shall include the following factors:

1. An overview of the evaluation model with a clear roadmap describing all parts of the evaluation model, their relationships with each other, and where they are located in the documentation;
2. A description of the analysed event scenario including:
    a) Facility initial conditions,
    b) The initiating event and all subsequent events and phases of the event analysed, and
    c) The physical phenomena, systems and component interactions, including human-machine interaction, influencing the outcome of the event;
3. A description of the code selection and assessment that includes:
    a) Code models,
    b) A description of each experimental and analytical test,
    c) Why each test was chosen,
    d) Success criteria,

e) Diagrams of the test facility and the location of the instrumentation for the experimental test, and

f) All code options used in the calculations;

4. A determination of the code uncertainty for a sample event calculation.

6.6 Safety Analysis Review and Update

6.6.1 Review of Safety Analysis Results

The licensee shall systematically review the safety analysis results to ensure that they are correct and meet the initial goal of the analysis. The results shall be assessed against the relevant requirements, applicable experimental data, expert judgment, comparison with similar calculations and sensitivity analyses.

The licensee shall review the analysis results using one or more of the following techniques, depending on the objectives of the analysis:

1. Supervisory review;
2. Peer review;
3. Independent review by qualified individuals; and
4. Independent calculations using alternate tools and methods to the extent practicable.

6.6.2 Update of Safety Analysis

6.6.2.1 Periodic Update

The safety analysis shall be updated within five years of the date that one was last conducted, unless otherwise approved in writing by the Commission or a person authorized by the Commission.

The updated safety analysis shall take into account the following elements:

1. The most up-to-date and relevant information and methods, including the experience gained and lessons learned from the situations, events, problems or other information pursuant to this regulatory standard;
2. Significant changes in facility configuration, conditions (including those due to aging) and operating parameters and procedures;
3. Research findings; and
4. Advances in knowledge and understanding of physical phenomena.

6.6.2.2 Unscheduled Update

In addition to periodic updates, the safety analysis shall also be updated following the discovery of information that may reveal a hazard to the health and safety of persons, security or the environment, that is different in nature, greater in probability, or greater in magnitude than was previously presented to the Commission in the licensing documents. Such information includes:

1. Major design changes or refurbishments, or both;
2. Changes due to new experiments; and
3. The occurrence of an event that was not considered in the safety analysis.

6.7 Quality of Safety Analysis

Safety analysis shall be subjected to a comprehensive QA program applied to all activities affecting the quality of the results. The QA program shall identify the quality assurance standards to be applied.

The QA program shall include documented procedures and instructions for the complete safety analysis process, including, but not limited to:

1. Collection and verification of facility data;
2. Verification of the computer input data;
3. Validation of facility and analytical models;
4. Assessment of results of simulations; and
5. Documentation of analysis results.

GLOSSARY

Acceptance criteria
Specified bounds on the value of a functional or condition indicator used to assess the ability of a system, structure or component to perform its design function.

Accident
Any unintended event, including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.

Anticipated operational occurrence (AOO)
An operational process deviating from normal operation that is expected to occur once or several times during the operating lifetime of the facility but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety nor lead to accident conditions.

Beyond design basis accident (BDBA)
Accident conditions less frequent and more severe than a design basis accident. A BDBA may or may not involve core degradation.

Common cause
A cause for a concurrent failure of two or more structures, systems or components, such as natural phenomena (earthquakes, tornadoes, floods, etc.), design deficiency, manufacturing flaws, operation and maintenance errors, human-induced destructive events and others.

Design basis
The entire range of conditions for which a facility is designed in accordance with established design criteria, and for which damage to the fuel and release of radioactive material are kept within authorized limits.

Design basis accident (DBA)
Accident conditions against which a facility 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.

Deterministic safety analysis
An analysis of facility responses to an event, performed using predetermined rules and assumptions (e.g., those concerning the initial facility operational state, availability and performance of the facility systems and operator actions). Deterministic analysis can use either conservative or best estimate methods.

Event category
A group of events characterized by the same, or similar, cause and similarity in the governing phenomena.

Graded approach
A risk-informed approach in which application of the safety analysis requirements is commensurate with the risk associated with the facility.

Non-power reactor
A fission reactor designed for research and testing or isotope production, or both.

Normal operation
Operation of a facility within specified operational limits and conditions including startup, power operation, shutting down, shutdown, maintenance, testing and refueling.

Nuclear power plant (NPP)
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*.

Operational limits and conditions (OLC)
A set of rules setting forth parameter limits or conditions for the safe operation, limiting safety system settings, surveillance requirements, operational constraints in event of safety system outages, and administrative controls for all operating modes, and prescribe the details on technical guidelines.

Probabilistic safety assessment (PSA)
For a NPP or nuclear fission reactor, a comprehensive and integrated assessment of the safety of the plant or reactor. The safety assessment considers the probability, progression and consequences of equipment failures or transient conditions to derive numerical estimates that provide a consistent measure of the safety of the plant or reactor, as follows:

1. A Level 1 PSA identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and massive fuel failures;

2. A Level 2 PSA starts from the Level 1 results, and analyses the containment behaviour, evaluates the radionuclides released from the failed fuel and quantifies the releases to the environment; and

3. A Level 3 PSA starts from the Level 2 results, and analyses the distribution of radionuclides in the environment and evaluates the resulting effect on public health.

A PSA may also be referred to as a Probabilistic Risk Assessment (PRA).

Safety function
A specific purpose that must be accomplished for safety.

Safety limit
A limit on process variables within which the reactor systems have been shown to be safe.

Safety system setting
The setpoint for a parameter at which protective safety system action will automatically shut down the reactor to prevent the corresponding safety limit from being exceeded.

Sensitivity analysis
A quantitative examination of how the behaviour of a system varies with change, usually in the values of the governing parameters.

Safety systems
Systems designed for the safe shutdown of the reactor, removal of the residual heat from the core, and prevention of release of radioactive material into the environment in case of an accident.

Single failure
A random failure that results in the loss of capability of a component to perform its intended safety function. Consequential failures resulting from a single random occurrence are considered to be part of the single failure. Failure of passive components can be excluded, provided they are shown to be designed, manufactured, installed, maintained and inspected to a recognized standard and are not adversely affected by the event being analysed.

Single failure criterion
The criterion used to determine whether a system is capable of performing its function in the presence of a single failure.

REFERENCES

1. *Nuclear Safety and Control Act* and regulations. Canadian Nuclear Safety Commission, Ottawa, 2000.

2. *General Nuclear Safety and Control Regulations*. Canadian Nuclear Safety Commission, Ottawa, 2000.

3. *Class I Nuclear Facilities Regulations*. Canadian Nuclear Safety Commission, Ottawa, 2000.

4. *Safety Requirements of Research Reactors*, DS 272, IAEA Safety Standards Series. International Atomic Energy Agency, 2004.

5. *Radiation Protection Regulations*. Canadian Nuclear Safety Commission, Ottawa, 2000.

6. *Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants*, CSA-N286.7-99. Canadian Standards Association, 2003.

7. Regulatory Guide G-149, *Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors.* Canadian Nuclear Safety Commission, Ottawa, 2000.

APPENDIX A – ACCEPTANCE CRITERIA

A.1 Safety Requirements

Table A.1 identifies the baseline safety requirements for AOO and DBA. Additional requirements may be applied to specific events. Acceptance criteria shall be identified for the practical application of these requirements.

Table A.1: Safety Requirements

#	Requirements	AOO	DBA	Notes
1	No reliance on safety systems	A	N/A	
2	No consequential degradation of fuel condition	A	N/A	In this requirement, degradation of fuel condition means that the fuel is no longer qualified for continuous use after being subjected to the predicted conditions.
3	Fuel configuration allows removal of residual heat	A	A	For events where the initiating event is in a single fuel channel or its associated piping, these requirements do not apply to that channel or the fuel associated with it.
4	No further fuel damage after long-term cooling system re-establishes adequate cooling	A	A	For events where the initiating event is in a single fuel channel or its associated piping, these requirements do not apply to that channel or the fuel associated with it.
5	No fuel breakup due to rapid energy addition	A	A	
6	Avoidance of prompt criticality	A	A	
7	Fuel channel configurations allows removal of residual heat	A	A	For events where the initiating event is in a single fuel channel or its associated piping, these requirements do not apply to that channel or the fuel associated with it.
8	No consequential failure of safety systems functions	A	A	

#	Requirements	AOO	DBA	Notes
9	No consequential loss of primary cooling system integrity	A	A	For events where the initiating event is in a single fuel channel or its associated piping, these requirements do not apply to that channel or the fuel associated with it.
10	Containment/confinement remains within design pressure range	A	A	
11	No consequential hydrogen explosion or deflagration in any facility system	A	A	
12	Reactor remains subcritical after shutdown	A	A	
13	Spent fuel remains subcritical	A	A	
14	Spent fuel cooling is maintained	A	A	

Note: Symbols “A” and “N/A” used in these columns signify “Applicable” and “Not Applicable” respectively.