Development of Canadian Seismic Design Approach and Overview of Seismic Standards

Aman Usmani[1] and Tarek Aziz [2]

[1] Current Chair of CSA N289 Technical Committee on Seismic Design, Amec Foster Wheeler, 4th Floor, 393 University Avenue, Toronto, Ontario M5G 1E6, Canada

[2] Past Chair of CSA N289 Technical Committee on Seismic Design, TSAziz Consulting Inc., Mississauga, Ontario L5L 3N8, Canada

Abstract

Historically the Canadian seismic design approaches have evolved for CANDU® nuclear power plants to ensure that they are designed to withstand a design basis earthquake (DBE) and have margins to meet the safety requirements of beyond DBE (BDBE). While the Canadian approach differs from others, it is comparable and in some cases more conservative.
The seismic requirements are captured in five CSA nuclear standards which are kept up to date and incorporate lessons learnt from recent seismic events.
This paper describes the evolution of Canadian approach, comparison with others and provides an overview and salient features of CSA seismic standards.

1. Introduction

Historically the Canadian seismic design approach [16] has evolved since the late sixties and early seventies for the CANDU® nuclear power plants to ensure that the plants are designed conservatively and are robust to withstand design basis earthquake (DBE) and have adequate margins to meet the safety requirements of beyond design basis earthquake (BDBE). This was achieved due to the foresight of the designers who incorporated good engineering practices in material selection, verification using simulated performance tests and good quality fabrication and construction. Importance of the robust SSC supports was recognized right from the outset. While the Canadian seismic approach differs in some aspects from others, it is comparable and in some cases is more conservative than other international approaches (for example see Table 1).

The seismic design and qualification requirements and procedures are captured in a series of five Canadian Standard Association (CSA) nuclear standards. These standards are prepared by the technical teams that include participation of industry wide stakeholders with expert knowledge and pragmatic approach to design. The CSA standards are kept up to date on a periodic basis and incorporate lessons learnt from seismic events such as Fukushima.

The current paper briefly describes the evolution of Canadian approach, comparison with others and provides an overview and their salient features of the current CSA seismic standards.

The references and bibliography [1] to [46] included in this paper can provide more details and background on which the foundation of current CSA standards and their subsequent development is based.

2. Early Days Prior to CSA Seismic Standards

Even before the formalization of seismic design methodology in the publication of the seismic CSA standards [1] to [5] in the early eighties, the CANDU designers had recognized the importance of the seismic loads and included consideration of lateral loads based on static load considerations in the safety related systems such as the reactor, containments and the heat transport systems. These static seismic loads and robust restraints in the safety related structures, systems and components (SSCs) were implemented in the Douglas Point, Pickering A, Bruce A and other earlier reactor designs. The benefit of these considerations was seen years later when these early plants were successfully assessed for seismic adequacy without much hardware change using Seismic Margin Assessment (SMA) methodology [1, 31, 42, and 46]. The detailed procedures for seismic design were incorporated in the form of Design Guides at AECL and Ontario Hydro (predecessor of Ontario Power Generation). These design guides became the starting point for development of the CSA seismic standards.

Another benefit of the seismic concepts implemented from the outset in the earlier reactor designs was that the subsequent advanced CANDU reactor designs which evolved based on these earlier designs did not have to make significant hardware changes to fully qualify the newer designs using the state of the art seismic qualification methods and the increased seismic demands.

Based on the earlier seismic design guides and internal procedures along with incorporating the then current knowhow around the world and especially in the USA, the Canadian Nuclear seismic standards were developed under the auspices of the Canadian Standard Association.

Along with the CSA standards [1 to 5], Canada adopted the use of the ASME boiler and Pressure Vessel Codes [12]. The ASME codes were developed for the light water reactors and did not address the specific design features such as the horizontal channel design involving pressure tubes made of zirconium material and connected to the stainless steel end fittings by rolled joint, small diameter feeder pipes made of carbon steel as part of the heat transport system and on power refueling of the CANDU design. The seismic approach methodology differences [16] included designing for only one level of earthquake (DBE or SDE) and no operating basis earthquake (OBE) as opposed to the two i.e. OBE and SSE in the USA. The seismic approach differences include seismic design basis earthquake (DBE) as ASME Level C as opposed to the safe shutdown earthquake (SSE) as ASME Level D and OBE as Level B in the USA. Canadian approach also requires fatigue consideration for the DBE. As well, there is no mandatory seismic trip in CANDU nuclear power plants. Addressing of these differences meant obtaining clarification and approvals from the Canadian Regulatory Authorities AECB (that subsequently became CNSC) and MCCR (that subsequently became TSSA) and embodying these in the design guides and procedures prior to publication of the CSA standards.

3. Development of Canadian Seismic philosophy and CSA Seismic Standards

CSA N289 Series of Standards [1 to 5] were initiated in the early seventies in response to the recognition on the part of the utilities and industries concerned with nuclear facilities in Canada of a need for consistent standards for seismic design and qualification of nuclear structures, systems, and components (SSCs) of nuclear power plants.

Early on it was recognized that although the standards includes regulatory requirements in addition to those of a technical nature, users of these Standards should recognize that they have the force of law only when adopted by the Canadian Nuclear Safety Commission (CNSC) or, in countries other than Canada, the appropriate regulatory body.

The CSA N289 series of Standards [1 to 5] sets forth the general requirements for seismic design and qualification of Nuclear Power Plants in Canada. The series applies to all structures, systems, and components (SSCs) of nuclear power plants requiring seismic qualification based on nuclear safety considerations. The Standards were developed for nuclear power plants in regions of low to moderate seismic hazard which is comparable to the levels near Canada’s existing nuclear facilities.Nuclear standards were primarily focused on power plants, but they also provide guidance on other types of nuclear facilities, including uranium mines and mills and other Class I nuclear facilities

CSA Program’s objectives are:

- To share industry knowledge by including historical perspective in documents and by giving younger technical personnel exposure to experienced veterans.
- Offer ongoing guidance to the industry as an alternative to regulatory documents
- Provide an expert panel to clarify standards through “Requests for Interpretation (RFI)”
- Provide standards and forums to support licensing and regulation

4. Applicability of CSA N289 Series of Standards

- The CSA N289 series of Standards [1 to 5] shall apply to:
- Containment structures classified as Class Containment in accordance with CSA N287.1
- Safety-related structures, systems, or components (SSCs), whose failure to perform the intended function will impede the safe operation and shutdown of the reactor;
- Electrical systems, control systems, and instrumentation whose components are required to demonstrate a safety function
- The balance of nuclear power plant SSCs, if an SSC is required to perform any safety function
- SSCs that are not required for nuclear safety but whose failure or dislocation could affect the safe operation of any safety-related system, the requirements of the CSA N289 series of Standards shall apply, at a minimum, to the supports, restraints, and anchors of such SSCs; and
- Temporary structures or installations whose failure or dislocation can result in contact with a seismically qualified SSC. Lightweight components or temporary structures that would not impair safety-related SSCs may be exempted from this requirement.

5. Earthquake Definitions used in Canada

The following Earthquake definitions are used in Canada [1]:

Design basis earthquake (DBE) — an engineering representation of potentially severe effects at the site due to earthquake ground motions having a selected probability of exceedance of 1×10^{-4} per year, or such probability level as determined by the regulatory authority.

Notes: (1) DBE ground motions are usually referred to as an "earthquake", and can take the form of a response spectrum, or time history of acceleration, velocity, or displacement. **(2)** The DBE is used for the seismic qualification of certain structures, systems, and components. It is used as an input for nuclear power plant seismic design, analysis, and testing to produce a design that is adequate for the specified seismic hazard. **(3)** The DBE for some older plants was based on an estimated probability of exceedance of $1x10^{-3}$ per year or was established deterministically (i.e., without probabilistic measures)**(4)** The free-field design ground response spectrum shall be specified at an annual frequency of 1×10^{-4} or lower, at the statistical mean confidence level as a minimum.

Checking/review level earthquake (CLE) — an engineering representation of earthquake ground motion chosen to have a lower probability of exceedance than the design basis earthquake (DBE).

Notes: (1) The CLE is used to identify any nuclear safety-related SSCs that might have insufficient seismic ruggedness, ductility, or inelastic response capability to withstand earthquakes exceeding the DBE. **(2)** The CLE is usually applied to existing plants, or existing designs of proposed plants, as part of seismic evaluation. **(3)** The probability of exceedance of the CLE is generally agreed upon by the owner/licensee and the regulatory authority. A probability of exceedance of 1×10^{-4} to 1×10^{-5} per year is typically selected.

Site operating earthquake (SOE) — an engineering representation of ground motion established using the best estimate of the 1×10^{-2} seismic hazard ground motions.

Notes: (1) The SOE represents a common earthquake, which is most likely to be experienced by a nuclear plant over its operating lifetime. **(2)** Although the SOE is similar to the operating basis earthquake (OBE), there are no corresponding regulatory actions required if the SOE is exceeded. **(3)** The SOE is considered the target level for which there is no effect on normal plant operations.

6. Salient Features of CSA N289.1

N289.1 [1] provides the general requirements for seismic design and qualification of nuclear power plants. It provides guidelines for identifying structures and systems requiring seismic qualification based on nuclear safety considerations.

Two seismic categories are used to identify the extent to which SSCs shall remain operational during and/or after an earthquake: (a) Seismic Category A includes those SSCs that shall maintain their structural integrity and shall retain their pressure boundary integrity (where applicable) during and/or following an earthquake. (b) Seismic Category B includes those SSCs that shall maintain their structural integrity and detailed functional requirements during and/or following an earthquake. Category B SSCs shall also retain their pressure boundary integrity, where applicable. Nuclear power plant SSCs may be seismically qualified using analysis (as described in CSA CAN3-N289.3) [3] or other methods, as appropriate.
CSA N289.1 [1] covers Seismic Margin Assessment (SMA) [1, 31, 42, and 46] and Seismic Probabilistic Safety Assessment (SPSA) Methodologies [1, 27 to 30, and 44]. The following is a summary of these methodologies:

6.1 Seismic Margin Assessment (SMA) Methodology

SMA methodology [1, 42, and 46] provides a means of quantifying the seismic capacity of SSCs required to perform essential safety functions during and following an earthquake. This methodology expresses seismic capacity in terms of a high confidence (95%) of low probability (5%) of failure (HCLPF) relative to a CLE. The CLE is selected at a lower probability of exceedance than the plant DBE in order to verify that SSCs have capacity to safely withstand earthquake ground motions larger than the DBE. A success path of SSCs necessary to safely shut down, cool, contain, and monitor the plant following an earthquake shall be selected. The success path shall be consistent with plant operator training and procedures, reactor safety, design, and system engineering interests. Alternatively, event tree/fault tree modelling of essential safety and support systems, similar to a seismic probability safety assessment (SPSA), may be used. Experienced and trained engineers shall perform walkdown evaluations of success path SSCs using evaluation criteria developed from documented earthquake experience and generic qualification and fragility test data to enable potential seismic interactions to be identified and as-built conditions. The SMA methodologies along with limitations specified for the use of these methodologies are intended to assist in the identification of the HCLPF of success path items and weak links.

6.2 Seismic Probabilistic Safety Assessment (SPSA) Methodology

The SPSA methodology [1, 32 and 44] models the plant response to initiating events using fault trees and event trees. The conditional probability of failure of essential SSCs is represented by fragility curves. Using the event tree/fault tree models, fragility curves, and probabilistic seismic hazard curves, the frequency of core damage can be calculated. An SPSA is generally performed by building on and modifying internal event probabilistic safety assessment models. The internal event and the fault trees are modified to reflect sseismically induced failures/conditions (e.g., seismic spatial interactions); the seismically induced failure of passive items such as structures and supports; and the common-cause effects of seismic excitation, seismically induced relay chatter, and operator error probability under seismically affected environments.

6.3 Seismic Evaluation of Existing Plants

Specific seismic analysis, test data, or observed performance of SSCs in heavy industrial facilities subjected to major earthquakes, as well as generic seismic testing information, forms the basis for seismic capacity determinations for components commonly found in nuclear power plants [46]. The SMA and SPSA methodologies have been developed to establish the seismic margin of nuclear power plant SSCs relative to the seismic hazard at a plant site. Seismic evaluation of existing nuclear power plant SSCs can be required when llicence conditions change; current seismological and geological data indicates that the site seismic hazard has changed; or mmajor modifications, replacements, or additions to the plant are made.

In addition CSA N289.1 [1] covers Operational Requirements for seismic events such as the ooperator response to seismic events, post-seismic recovery and shut down and required operator actions [5].

7. Salient Features of CSA N289.2

Ground motion determination for seismic qualification of nuclear power plants is covered in CSA N289.2 [2] along with the investigations for seismically induced phenomena. Determination of ground shaking hazard involves the following aspects: 1) geological and seismological investigations performed for site, site vicinity, and region;2) Development of seismic source models;3) Evaluation of seismogenic potential;4) development of earthquake recurrence models;5) evaluation of maximum potential earthquake; 6) development or adoption of ground motion prediction equations;7) execution of a probabilistic seismic hazard assessment; 8) generation of uniform hazard response spectra; 9)Development of scenario earthquakes; 10) evaluation of seismic hazard uncertainties; and 11) evaluation of seismically induced phenomena (tsunami, seiche, volcanism, slope instability, surface faults, surface instability, and dam failures) to be accommodated by nuclear power plant siting and design.

A procedure for the determination of seismic ground motion is described in CSA N289.2 [2] that is generally compatible with procedures described in similar guides (e.g., IAEA Safety Standards Series No. NS-G-3.3). Background for the CSA N289.2 procedures is available in References [20 to 30].
.CSA N289.2 establishes the basis for a family of seismic hazard results that can be used as input to CSA N289.3 [3].The Standard was developed for the determination of ground motions for regions of low to moderate seismic hazard, comparable to the levels near Canada's existing nuclear power plants. Guidance regarding additional provisions for high seismic hazard sites may be obtained from IAEA Safety Standards Series No.NS-G-3.3.

8. Salient Features of N289.3

CSA N289.3-10 [3], Design procedures for seismic qualification of nuclear power plants provides design requirements and methods (i) for determining the engineering representation of ground motion, ground response spectra, and floor response spectra for use in the design and seismic qualification of SSCs; and (ii) for performing seismic qualification of specified SSCs by analytical methods; The Standard applies to structures, systems, and components (SSCs) in nuclear power plants that require seismic qualification by analytical methods (see CSA N289.1 [1]). This Standard may also be applied to SSCs that might not require explicit seismic qualification as deemed appropriate by the owner/licensee or by authorities having jurisdiction [1] and as appropriate, to other nuclear facilities under the jurisdiction of the Nuclear Safety and Control Act.
Design intent reflected in this standard is to follow seismic qualification process to demonstrate that seismically qualified SSCs have the capability to safely withstand the effects of the plant site seismic hazard.

8.1 Design ground motion

Seismic design of nuclear power plant SSCs shall be based on DBSGMs representing the seismological, geological, and geotechnical conditions of the site, expressed in a form applicable to dynamic analysis of the plant SSCs. Seismic design of nuclear power plant SSCs shall be performed using standard-shape ground response spectra, or site-specific spectra that have been conservatively defined to account for uncertainties in the site seismic hazard. The design ground response spectra shall be based on the free-field ground motion or reference ground condition modified to incorporate the site-specific geological conditions. The minimum design horizontal response spectra used in the design of new nuclear power plant SSCs shall be (a) the standard-shape ground response spectrum anchored to a peak ground acceleration of 0.1 g on rock and (b) modified to take into account the site-specific geological conditions.

8.2 Seismic qualification by analytical methods

The N289.3 standard specifies acceptable methods of dynamic analysis for the seismic qualification of safety-related SSCs in nuclear power plants and defines technical requirements for applying these methods. Structures that support safety-related systems shall undergo dynamic analysis to predict the responses of a system when subjected to the design basis ground motion.

8.3 Seismic fatigue analysis

In general, seismic fatigue analysis [3, 13] of structures is not required due to the limited number of stress cycles imposed during a seismic event. For components and component supports, which require fatigue analysis, the fatigue analysis shall be in accordance with requirements provided in the standard.
Seismic fatigue analysis of ASME Class 1 components and supports shall not be required when the range of primary plus secondary stresses due to the seismic load alone is limited to 3 Sm (the design stress intensity) or equivalent. The minimum number of seismic cycles shall be 25 at the maximum peak response levels when determining the equivalent seismic fatigue effects from the design fatigue curves of Figure I-9.0 of the ASME BPVC, Section III, and Division 1 [12].

8.4 Other seismically induced phenomena

Additional effects of seismically induced phenomena, such as tsunami, seiche, volcanism, or dam failure shall be mitigated by siting, layout, and design of the nuclear power plant SSCs. Flooding can be induced by ground shaking or as a consequence of the earthquake effects

8.5 Documentation

Seismic qualification by analysis shall be documented to demonstrate that the structure or equipment has met its design requirements under the specified design seismic environment.in a structured format that can readily be audited by the appropriate authority having jurisdiction.

9. Salient Features of N289.4

The CSA CAN3-N289.4, Testing procedures for seismic qualification of CANDU nuclear power plants [4] provides design requirements and methods for seismic qualification of specific components and systems by testing methods; for SSCs requiring seismic qualification

9.1 Specification development

This Standard is intended to provide a basis for developing specifications for seismic qualification of SSCs by testing, or by a combination of analysis and testing, and to aid component purchasers, suppliers, and testing laboratories.

9.2 Acceptable methods and requirements for testing

This Standard presents several acceptable methods with the intent of permitting the user to make a judicious selection from among the various options. To meet the objectives of the standard, the seismic testing shall be performed in a manner that will demonstrate dynamic response characteristics and acceptability of the test specimen to withstand and maintain its function as required during the expected level of shaking.

9.3 Testing of high frequency-sensitive components

When applicable, the test specification shall include provisions for testing of components [38 to 40] known to be sensitive to high frequencies (e.g., relays, contactors, switches, potentiometers, sensors, circuit breakers, and microprocessors). I

9.4 Simulating input motions and accounting for aging

The shake table motion shall simulate the postulated earthquake event in as realistic a manner as possible using one of the following forms: required response spectrum (RRS); required input motion (RIM); required time history; or required power spectral density. Coincident loading and aging degradation effects shall be taken into account.

9.5 Application of qualification by similarity

Components similar to the one that has been qualified to a given seismic motion are not required to be retested.

9.6 Seismic qualification by combined testing and analysis

Certain components cannot be practically qualified by analysis or testing alone. This may be due to the size of the component, its complexity, or the extent of variability of subcomponents within a larger component.

9.7 Testing facility qualification

All activities related to seismic testing shall be performed by qualified personnel using qualified testing equipment and approved procedures.

9.8 Test specification and Test Plan

The purchaser shall provide the test specification prepared prior to the qualification test(s) and shall describe the test activities and requirements. The supplier shall submit a test plan, including the test sequence and procedure, to the purchaser for review and approval before the test can proceed

9.9 Qualification report

The equipment supplier, in collaboration with the testing facility, shall provide a qualification report to properly document and to clearly demonstrate the compliance with the test specification.

10. Salient Features of N289.5

CSA N289.5 [5] deals with Seismic instrumentation requirements for nuclear power plants and nuclear facilities and establishes the requirements for seismic instrumentation. This Standard provides a basis for specifying requirements for seismic instrumentation. The recorded data of seismic activity will be used to compare actual and predicted responses, and to assess the need for further detailed inspections. This Standard aids owners of nuclear power plants and nuclear facilities in the determination of the extent and nature of instrumentation to be installed. It also is intended to aid owners and equipment suppliers by specifying instrumentation commensurate with Canadian nuclear safety principles.

11. Summary and Conclusions

The Canadian nuclear plants have been designed using sound engineering practices even before the development of the CSA seismic standards.

The CSA standards [1 to 5] are prepared and kept up to date by a technical committee with representation from all nuclear industry stakeholders under the auspices of the CSA. The suite of 5 seismic standards [1 to 5] capture all aspects of safe and sound seismic design and are kept updated and harmonized with international standards.

In terms of requiring qualification for only one earthquake using level C limits of ASME section III code and requiring seismic fatigue consideration, no mandatory seismic trip, the Canadian seismic approach is somewhat different and more conservative than other practices.

Canadian standards have been considering beyond design basis for the last 30 years as additional defense in depth (e.g. CLE for seismic design and UPC for containment design).

Conservative stress limits and design requirements imposed on the treatment of the design basis can ensure adequate design margins for beyond design basis.

Civil structures and mechanical systems typically have an inherent margin of 2 to 3 if the Canadian rules of design are followed (e.g. Level C emergency conditions, fatigue analysis for DBE, stress limits for structures)

Table 1: Comparison of Canadian and USA Seismic Design Practices

		Canada	U.S.A
1	Design Earthquakes	DBE(deterministic/probabilistic SDE(one in 100 years) NBCC	SSE (Deterministic) OBE=1/2 SSE
2	GRS Shape	CAN3-N289.3 Based on Acceleration, velocity and Displacement Velocity normalized 90 percentile*	RG1.60 Based on Acceleration only Acceleration normalized 84 percentile
3	Damping Values	Per CSA N289.3 (generally lower)*	Per RG 1.61
4	Seismic Classification	DBE-qualified SDE-qualified NBCC-qualified	Category I Category II Category III
5	Seismic Qualification Methods & Approach	Similar Fatigue includes*	Similar Fatigue for SSE not included
6	Allowable ASME Stress Levels	Level C (lower) *	Level D for SSE
7	Shutdown Systems	Two Qualified *	One Qualified
8	Loading Combinations	DBE+ Pressure(reduced LOCA)	SSE+ LOCA (full)
9	Containment Design	CSA N287 Series	ASME-Division 2, NRC SRP3.8.1
10	Other Structures	A23.3, S16, S16.1, CISC	ACI 349, NRC SRP 3.8.3 and 3.8.4, AISC

*More Conservative

12. References & Bibliography

[1] CAN3-N289.1, General Requirements for Seismic Qualification of CANDU Nuclear Power Plants

[2] CAN3-N289.2, Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants

[3] CAN3-N289.3, Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants

[4] CAN3-N289.4, Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants

[5] CAN3-N289.5, Instrumentation, Inspection and Records for Seismic Qualification of CANDU Nuclear Power Plants

[6] Mohraz, B., W.J. Hall and N.M. Newmark. A Study of Vertical and Horizontal Earthquake Spectra. Nathan M. Newmark, Consulting Engineering Services, Urbana, Illinois, USAE C Contract No. AT (49-5)-2667, 1972.

[7] Tsai, N.C., "Spectrum–Compatible Motions for Design Purposes", Journal of the Engineering Mechanics Division, ASCE, April, 1972.

[8] Ad Hoc Group on Soil-Structure Interaction, "Analysis for Soil-Structure Interaction Effects for Nuclear Power Plants", Nuclear Structures and Materials Committee of the Structural Division of ASCE, 1976.

[9] Aziz, T.S. and C.G. Duff, "Decoupling Criteria for Seismic Analysis of Nuclear Power Plant Systems", ASME/CSME Pressure Vessels and Piping Conference with Nuclear and Materials Division, June 25-29, Montreal, Canada.

[10] "Structural Analysis and Design of Nuclear Plant Facilities", ASCE Manuals and Reports on Engineering Practice, No. 58, 1980.

[11] US Regulatory Guide 1.92, "Combining Model Responses and Spatial Components in Seismic Responses Analysis", February, 1976.

[12] ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components Division 1.

[13] Duff, C.G. and A.C. Heidebracht, "Earthquake Fatigue Effects in CANDU Nuclear Power Plant Equipment", Proceedings, 3rd Canadian Conference on Earthquake Engineering, Montreal, June 4-6, 1979.

[14] Aziz, T.S. and J.K. Biswas, "Spectrum-Compatible Time-Histories for Design of Nuclear Power Plants", Proceedings, 3rd Canadian Conference on Earthquake Engineering, Montreal, June 4-6, 1979.

[15] Shinozuka, M. and T. Koike. 1979. Estimation of Structural Strains in Underground Lifeline Pipes. Lifetime Earthquake Engineering — Buried Pipelines. Seismic Risk and Instrumentation. PVP-34, ASME Press, pp. 31–48.

[16] Usmani, S.A., “Seismic Qualification of CANDU-PHW NPP Equipment” Nuclear Journal of Canada, Vol. 1, No. 3 1987 September.

[17] Stevenson, J.D. and Coman O. 2006. Design of Hazardous Mechanical Structures Systems and Components for Extreme Loads. ASME Press.

[18] Tsai, N.C. 1972. Spectrum-compatible motions for design purposes. Journal of the Engineering Mechanics Division. ASCE.

[19] Usmani, S.A. and P.D. Baughman. 1996. Recommendation for Damping and Treatment of Modeling Uncertainty in Seismic Analysis of CANDU Nuclear Power Plant. 10th Symposium of Seismic Engineering, ASME PVP and ICPVT Conference. Montréal

[20] Adams, J. and G. Atkinson (2003). Development of seismic hazard maps for the proposed 2005 edition of the National Building Code of Canada. Canadian Journal of Civil Engineering Special Issue, 30(2), 255–271.

[21] Cornell, C. (1968). Engineering seismic risk analysis. Bull Seism. Soc. Am., 58, 1583–1606.

[22] Cornell, C. (1971). Probabilistic analysis of damage to structures under seismic loads. In Dynamic Waves in Civil Engineering, ed. D. Howells, I. Haigh, and C. Taylor. Wiley-Interscience, London.

[23] Earthquake Engineering Research Institute (EERI) Committee on Seismic Risk (1989). The basics of seismic risk analysis. Earthquake Spectra, 5, 675–702.

[24] Fenton, C.H., J. Adams, and S. Halchuk (2006). Seismic hazards assessment for radioactive waste disposal sites in regions of low seismic activity. Geotechnical and Geological Engineering, 24(3), 579–592.

[25] Halchuk, S., J. Adams, and F. Anglin (2007). Revised Deaggregation of Seismic Hazard for Selected Canadian Cities. 9th Canadian Conference on Earthquake Engineering, Paper 1188, 13 pp.

[26] McGuire, R. (1977). Seismic design spectra and mapping procedures using hazard analysis based directly on oscillator response. Intl. J. Earthq. Eng. Struct. Dyn., 5, 211–234.

[27] McGuire, R. (1995). Probabilistic seismic hazard analysis and design earthquakes: Closing the loop. Bull. Seism. Soc. Am., 85, 1275–1284.

[28] McGuire, R. (2004). Seismic Hazard and Risk Analysis. EERI Monograph MNO-10. Earthquake Engineering Research Institute, Oakland, CA.

[29] Senior Seismic Hazard Analysis Committee (SSHAC), R.J. Budnitz, G. Apostolakis, D. Boore, L. Cluff, K. Coppersmith, A. Cornell, and P. Morris (1996). Probabilistic seismic hazard analysis: A consensus methodology. U.S. Department of Energy, U.S. Nuclear Regulatory Commission, Electric Power Research Institute.

[30] Senior Seismic Hazard Analysis Committee (SSHAC). (1997). Recommendations for probabilistic seismic hazard analysis: Guidance on uncertainty and use of experts. U.S. Nuclear Regulatory Commission Report, NUREG/CR-6372.

[31] AECB Project No. 2.209.1 (1992) Pickering Seismic Safety Margin — Methodology

[32] EPRI 1002989 Seismic Probabilistic Risk Assessment Implementation GuideFinal Report, December 2003EPRI (Electric Power Research Institute)

[33] EPRI1007896 (2003) Seismic Evaluation Guidelines for HVAC Duct and Damper Systems

[34] EPRI1012023 (2005) Experience-Based Seismic Verification Guidelines for Piping Systems

[35] 1012967 (2005) Program on Technology Innovation: Effect of Negligible Inelastic Behavior on High Frequency Response

[36] 1013504 (2006) Program on Technology Innovation: Effect of Seismic Wave Incoherence on Foundation and Building Response

[37] EPRI 1014099 (2006) Program on Technology Innovation: Use of Cumulative Absolute Velocity (CAV) in Determining Effects of Small Magnitude Earthquakes on Seismic Hazard Analyses

[38] EPRI 1014379 (2006) Program on Technology Innovation: Sensitivity of Performance-Based Approaches for Determining Safe Shutdown Earthquake (SSE) Ground Motion for New Plant Sites

[39] EPRI 1015108 (2007) Program on Technology Innovation: The Effects of High-Frequency Ground Motion on Structures, Components, and Equipment in Nuclear Power Plants

[40] EPRI 1015109 (2007) Program on Technology Innovation: Seismic Screening of Components Sensitive to High-Frequency Vibratory Motions

[41] EPRI NP-5930 (1988) A Criterion for Determining Exceedance of the Operating Basis Earthquake

[42] EPRI NP-6041-SLR1 (1991) A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)

[43] EPRI TR-100082-T1 (1991) Standardization of the Cumulative Absolute Velocity

[44] EPRI TR-103959 (1994) Methodology for Developing Seismic Fragilities

[45] EPRI TR-105849 (1996) Generic Seismic Technical Evaluations of Replacement Items for Nuclear Power Plants — Item-Specific Evaluations

[46] SQUG (Seismic Qualification Utility Group) Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, 2001