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DEVELOPMENT OF THE ALTERNATE PRESSURIZED THERMAL SHOCK RULE (10 CFR 50.61a) IN THE UNITED STATES

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In the early 1980s, attention focused on the possibility that pressurized thermal shock (PTS) events could challenge the integrity of a nuclear reactor pressure vessel (RPV) because operational experience suggested that overcooling events, while not common, did occur, and because the results of in-reactor materials surveillance programs showed that RPV steels and welds, particularly those having high copper content, experience a loss of toughness with time due to neutron irradiation embrittlement. These recognitions motivated analysis of PTS and the development of toughness limits for safe operation. It is now widely recognized that state of knowledge and data limitations from this time necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical of the PTS Rule, 10 CFR 50.61. To remove the unnecessary burden imposed by these conservatisms, and to improve the NRC's efficiency in processing exemption and license exemption requests, the NRC undertook the PTS re-evaluation project. This paper provides a synopsis of the results of that project, and the resulting Alternate PTS rule, 10 CFR 50.61a.

KEYWORDS : Pressurized Thermal Shock, Nuclear Reactor Pressure Vessel, Embrittlement

1. INTRODUCTION

In the early 1980s, attention focused on the possibility that PTS events could challenge the integrity of the RPV because operational experience suggested that overcooling events, while not common, did occur, and because the results of in-reactor materials surveillance programs showed that US RPV steels and welds, particularly those having high copper content, experience a loss of toughness with time due to neutron irradiation embrittlement. These recognitions motivated analysis of PTS and the development of toughness limits for safe operation. It is now widely recognized that state of knowledge and data limitations from this time necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis [1] of the PTS Rule [2]. To remove the unnecessary burden imposed by these conservatisms, and to improve the staff's efficiency in processing exemption and license exemption requests, the NRC undertook the PTS re-evaluation project [3,4].

The PTS re-evaluation project was conducted between 1998 and 2009 by the United States Nuclear Regulatory Commission USNRC. Assistance and data was provided by the commercial nuclear power industry operating under the auspices of the Electric Power Research Institute (Electric Power Research Institute). Toward the end of this time the project findings were reviewed by the Advisory Committee on Reactor Safeguards (ACRS), the Nuclear Energy Institute (NEI), the public, and a panel of national and international experts. These reviews provided the basis for numerous model corrections and improvements. Based on the findings of this project, the NRC initiated rulemaking on a voluntary alternate to 10 CFR 50.61 in 2006 [5,6]. Rulemaking was completed on January 4, 2010 when 10 CFR 50.61a was published in the Federal Register [7].

This description of the PTS re-evaluation project begins in Section 2 with a discussion of the risk limits that provide the basis for the embrittlement-based screening limits adopted in 10 CFR 50.61a. Then Section 3 describes the probabilistic model that was used to develop relationships between risk limits and embrittlement limits. Section 4 describes the results obtained from this model, and Section 5 describes how they were used to establish regulatory limits for 10 CFR 50.61a. Section 6 compares the regulatory provisions of 10 CFR 50.61 and to 10 CFR 50.61a.

2. REGULATORY LIMITS

In the Atomic Energy Act of 1954, which allowed the first large scale commercial use of nuclear energy in the United States, the United States Congress instructed the Atomic Energy Commission (AEC, the precursor to the NRC) to "*provide adequate protection to the health and safety of the public*" from radiological hazards. In the years that followed some studies attempted to define quantitatively the level of risk that nuclear generation of electricity posed to the public, and to provide some rationalization regarding what risk levels could be regarded as acceptable [8,9]. Nevertheless, between 1954 and the late 1980s the methods used by the AEC, and later by the NRC, to ensure the "adequate protection" required by their legislative mandate were, by and large, those common to "deterministic" engineering analysis, i.e. bounding approaches, margins, and the use of the defense-in-depth principle. This situation began to change in the early 1980s. Motivated by the recommendations of the President's Commission on the Accident at Three Mile Island [10], and enabled by improvements in computational technology and PRA methodologies, the NRC pursued much more vigorously the formal definition of both qualitative and quantitative safety objectives, or "safety goals." It should also be added that having clearly articulated safety goals, along with accepted methods by which the performance of plants (or fleets of plants) relative to these goals can be measured, removes arbitrariness from, increases the transparency of, and improves the uniformity of the regulatory decision-making process.

The work on safety goals begun after the Three Mile Island accident culminated in the issuance of a safety goal policy statement in 1986 [11]. This statement, along with other policies that lead the way to the risk limits adopted in the PTS re-evaluation project, are illustrated in Fig. 1. Key points that link the PTS risk limits to Commission policy may be summarized as follows. The 1986 safety goal policy statement defines risk limits for plant operation in terms of quantitative health objectives (QHOs) that measure the prompt fatality risk to individuals, and the latent cancer fatality risk to society [11]. The QHOs for both limit the health and safety risk arising from nuclear plant operations to a very small fraction (< 0.1%) of the total public risk. In 2000, the 1986 policy statement was modified to include a subsidiary limit on the core damage frequency (CDF) of 1×10^{-4}/ry, and was clarified by stating that both the CDF and QHO limits were intended to guide generic agency decisions (e.g., rulemaking) [12]. The information in both policy statements was incorporated into, and augmented by, the publication of Regulatory Guide 1.154 [13]. This guide provided yet another subsidiary goal, the large early release frequency (LERF). RG1.154 also defined limits on the total CDF and LERF, as well as on the CDF and LERF seen to arise from any single cause (see the table in Fig. 1). Finally, this ΔLERF limit of 10^{-6}/ry was used in the PTS re-evaluation project to establish a limit on the through-wall cracking frequency (TWCF) of 10^{-6} events per reactor operating year. The TWCF limit is based on a conservatively assumed equivalence between TWCF and LERF [3].

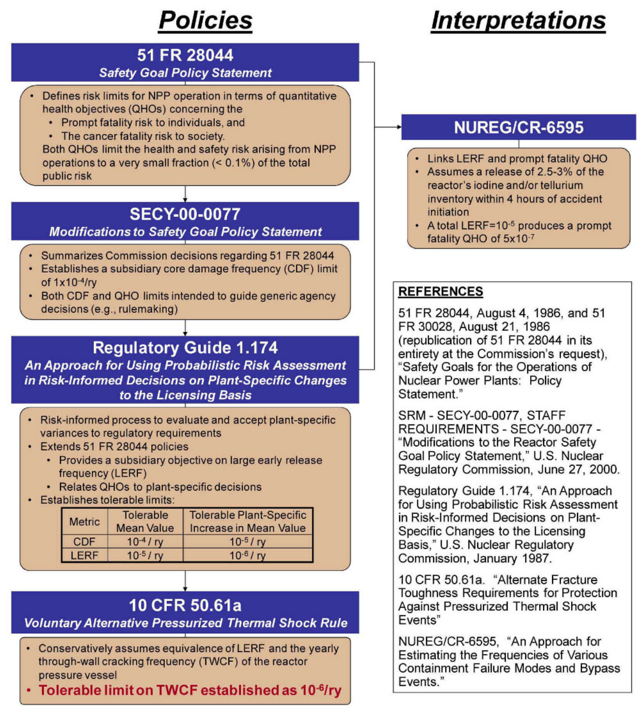

Fig. 1. Origins of Risks Limits for Nuclear Power Plant Operation in the USA.

Beyond the considerations that led to the 10^{-6} limit, the staff also considered the definition of vessel "failure." Failure was defined as the initiation of a rapidly propagating fracture from a pre-existing flaw in the vessel beltline region, followed by sufficient extension of that flaw to penetrate fully the thickness of the RPV wall. This definition was adopted because through-wall cracking of the RPV was viewed as a measure closely related to the potentially significant public health consequences that are discussed in Commission policy guidance. An assessment of the sequence of events between vessel "failure" and either core damage or LERF revealed that LERF is an unlikely consequence of through-wall cracking in the overwhelming majority of scenarios, thereby validating the conservatism of assuming that LERF=TWCF for the purpose of associating reference temperature based screening metrics with a numeric risk limit.

3. TECHNICAL MODEL / METHODOLOGY

3.1 Overview

Fig. 2 illustrates our overall model of PTS, which involves three major components:

1. <u>*Probabilistic Evaluation of Through-Wall Cracking Frequency*</u>: Estimates the frequency of through-