The Use of Probabilistic Safety Techniques for Evaluating the Advanced CANDU Reactor (ACR)

Michael D. Muhlheim/Oak Ridge National Laboratory (ORNL)

Donald A. Copinger/ORNL

Joseph W. Cletcher, II/ORNL

Mark A. Linn/ORNL

Donald L. Williams, Jr./ORNL

John N. Ridgely/U.S. Nuclear Regulatory Commission (NRC)

ABSTRACT THAT WAS SUBMITTED

The U.S. Nuclear Regulatory Commission (NRC) is anticipating licensing applications for reactor facilities that are significantly advanced beyond the current generation of operating reactors. One proposed reactor design, developed by Atomic Energy of Canada, Limited (AECL), is an Advanced CANada Deuterium Uranium (CANDU) Reactor (ACR), the ACR-700. The ACR is an enhanced version of earlier CANDU designs. However, unlike the CANDU reactors, which are heavy-water cooled and moderated reactors, the ACR-700 is a light-water cooled and heavy-water moderated reactor. In preparation of a possible design certification review, the NRC (with the assistance of ORNL) began examining selected areas of nuclear safety, identifying accidents that could potentially dominate the risk profile of the ACR-700 design, and evaluating other risk-important design and technology issues. This effort supports the NRC's policy that encourages the use of probabilistic risk assessment (PRA) in all regulatory matters. In addition to identifying potential initiating events and systems judged to be "important" to preventing and mitigating possible accident conditions, a prototype risk evaluation model for the ACR-700 was developed using the SAPHIRE computer code.

INTRODUCTION

U.S. Nuclear Regulatory Commission (NRC) policy encourages early discussions (prior to a license application) between the NRC and potential applicants to offer licensing guidance and to identify and resolve potential licensing issues early in the licensing process. On June 19, 2002, Atomic Energy of Canada, Limited (AECL) requested a pre-application review of their ACR-700 reactor [Advanced CANada Deuterium Uranium (CANDU) Reactor] for licensing in the United States.

Because current NRC policy requires applicants to submit a probabilistic risk assessment (PRA) and encourages its use in all regulatory matters, it is necessary for the NRC to have the capability to independently review an applicant's PRA. The building of these capabilities requires becoming familiar with the plant and identifying the most likely initiating events (IEs) and the systems needed to mitigate those IEs. This review provides feedback, observations, identification where additional information would be needed, and identification of areas that must be addressed by AECL during the design certification phase to allow the NRC to complete its safety determination of the ACR-700 design.

ACR-700

The work in this project was specific to AECL's ACR-700 reactor duplicated in Figure 1 [1]. The ACR-700 design is based on the CANDU 6 and CANDU 9 designs. Being an advanced reactor design, the ACR-700 includes more passive features, fewer active systems and components, and fewer operator interactions and interventions. Some basic features in the ACR-700 include

- horizontal fuel channels (i.e., each comprised of a pressure tube surrounded by a calandria tube) in a conventional, heavy-water-moderated calandria vessel, similar to a CANDU;
- reactor coolant pumps, steam generators, and the piping in the reactor coolant system based on the CANDU 6 and CANDU 9 designs;
- two fully independent and diverse shutdown systems, similar to a CANDU;
- at-power fueling and the fuel handling system, similar to a CANDU;
- slightly enriched (2 wt. % ^{235}U) UO_2 fuel, compared to natural uranium in a CANDU;
- a light-water coolant system, rather than heavy-water coolant used in a CANDU;
- an elevated water system that provides backup passive decay heat removal for severe core damage accidents (although the heat removal system is passive, it requires multiple valves to open to operate),
- a negative full-core coolant-void reactivity, rather than the positive reactivity of a CANDU; and
- a more compact core configuration and plant layout compared to a CANDU.

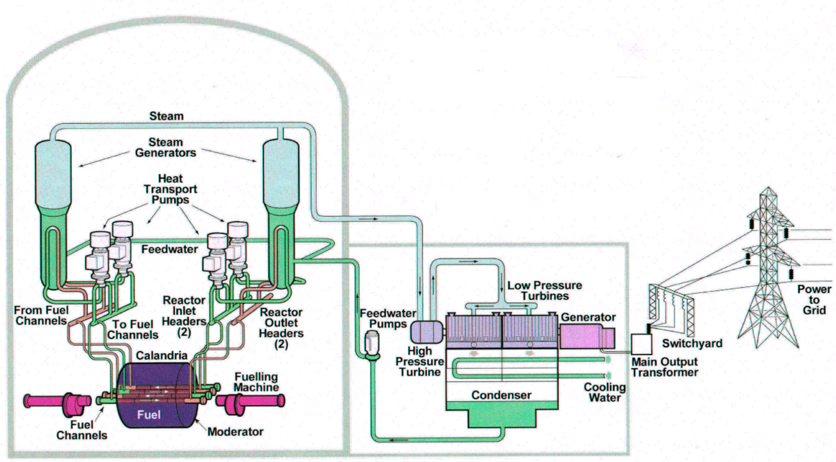

Figure 1. Features of ACR CANDU (*Source*: "Design Characteristics of ACR-700," *Twenty Third Annual Conference of the Canadian Nuclear Society*.)

TOOLS, EXPERTISE, AND DATA NEEDED FOR REVIEWS OF ADVANCED REACTOR PRAs

The research performed by the Oak Ridge National Laboratory (ORNL) for the NRC examined selected areas of nuclear safety, assisted in identifying accidents that could potentially dominate the risk profile of the ACR-700 design, and evaluated other risk-important design and technology issues. The research was focused on four general areas of study: (1) perform a critical review of existing PRA information regarding the ACR-700 design; (2) identify potential IEs for the ACR-700; (3) perform a high-level review of the ACR-700 systems to determine potential risk-significant design features; and (4) selection and use of the appropriate PRA review tools for application to the preliminary ACR-700 design. This latter task included the update of existing CANDU 3 models using the SAPHIRE code and the development of a prototype risk evaluation model that could be further developed to independently assess the ACR-700 design and supporting risk analyses.

PRA Documentation Review

One of the focus topics, or key areas identified for the ACR pre-application review was to review the ACR PRA methodology. Thus, selected generic CANDU (GC) and ACR probabilistic safety assessment (PSA) documents were reviewed to gain insights into the strengths and weaknesses of the PRA methodology and analysis used by AECL to support the design of the ACR-700. This work was performed with the understanding that any review of methodology and reference analysis documents can only yield insights into how the PRA will be performed; it is not a substitute for—nor can it be—an actual PRA review.

In identifying strengths and weaknesses of the methodology used in the GC PSA and ACR PSA, the guidance provided in NUREG/CR-2300, *PRA Procedures Guide* [2] and American Society of Mechanical Engineers (ASME) RA-S-2002, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications* [3] was considered. In addition, personal insights from the operational, design, and PRA experience of the ORNL staff conducting the review were included. General observations about the relative strengths and weaknesses (when compared to NUREG/CR-2300 and ASME RA-S-2002) of the methodology documents reviewed are provided below [4].

Reporting Strengths

- The PSA process appears to be consistent with typical PRAs for U.S. nuclear power plants (NPPs),
- the information provided on the PSAs is generally complete and easily understood, and
- the ACR PSA methodology report is more detailed and thorough than the GC PSA methodology report and clearly defines the scope of the PRA,

Reporting Weaknesses

- At times, there is a lack of sufficient details, supporting information, assumptions, or references in describing plant systems or the methodology being used. In addition, not all sources of information or assumptions are clearly identified or referenced. In some instances, a representative example or calculation is necessary to enhance the reader's understanding of the methodology.
- Eliminating typographical errors, calculational errors, or inconsistencies between tables, figures, text, and related reports is difficult because of the evolving nature of design stages; however, these errors reduce the credibility of the reports.
- The use of proprietary documents as references limits the understanding of assumptions, initial conditions, analyses, and operations that are critical to understanding the PRA.

Methodology Strengths

- PRAs completed within the last several years will be, in large part, based on NUREG/CR-2300 because ASME RA-S-2002 was not available prior to mid-2002. (Most individual plant examinations, or IPEs, for U.S. NPPs used NUREG/CR-2300 for guidance.) The ACR-700 PSA follows ASME RA-S-2002 and was updated with new or current methods when appropriate, thereby strengthening the credibility of the calculations and models (however, this practice was not always followed and, occasionally, out-of-date methods and data are used).
- Performing a PRA early in the design process not only helps in identifying design weaknesses, it also aids reviewers in understanding latent intricacies of the plant design and intended operations.
- AECL updated its component failure rates with recent plant data from its CANDU fleet of reactors. (Occasionally, 20B30 year old failure data or engineering judgment is used, see below.)

- The PRA uses 16-digit event identification nomenclature, which is helpful in interpreting the results because only the labels (not the full descriptions) are retained in the evaluation process.
- AECL is proactive in considering external events (i.e., flood, fire, and seismic events) early in the design.
- When compiling a list of potential IEs, AECL not only methodologically identifies those initiators that can lead to core damage, but those that can lead to a release of radioactive material.

Methodology Weaknesses

- Calculational errors and inconsistencies led to questioning of the accuracy of the calculations and validity of the results. In addition, not all of the modeling and analysis assumptions appear to be appropriate.
- The MAAP code (Modular Accident Analysis Program) has been accepted by the NRC for use with LWRs for severe accident analyses. AECL uses a β-version of a modified MAAP code for CANDUs─MAAP CANDU─that currently has not been reviewed by the NRC for its accident analyses.
- Not all components or systems appear to be considered [e.g., instrument air system (IAS) and the heating, ventilation and air-conditioning (HVAC) systems]. Even though the failure of many of the systems or components not considered may not increase the probability of core damage, others are important to safety in U.S. NPPs. The PSA needs to provide sufficient information to support their omission.
- The numerous weaknesses in the fire methodology, taken together, represent an important weakness.
- At times, the PSA models use 20B30 year old failure data. The validity of the data should be discussed, especially with respect to uncertainties.
- The data used to obtain generic β factors for its dependent failure analyses using the unified partial method (UPM) is 20B30 years old. In addition, because many IPEs for U.S. NPPs use the multiple greek letter (MGL) or α-factor method, AECL should fully document the implementation of the UPM and discuss its limitations in its PSA (e.g., the UPM does not address staggered testing, has limits on redundancy, etc.).

Identification of Initiating Events

A key element in the safety of any NPP design is the evaluation of the plant's ability to withstand events that could reasonably be postulated to occur and, if unmitigated, could lead to damage of the nuclear fuel and/or release of radionuclides to the atmosphere. The philosophy of designing NPPs to withstand these potential accidents has been a fundamental building block in the safety assurance of U.S. and non-U.S. reactor designs for many years. One of the first steps to ensure that the reactor design is sufficiently robust to withstand these accidents is to identify a comprehensive list of IEs that might lead to fuel damage or radionuclide releases. A common method (and one employed by U.S. NPPs) for providing a reasonably complete list of potential IEs for a PRA is to: [3]

- begin with an existing list of known initiators,
- group the IEs so that the events in the same group have similar mitigation requirements to facilitate an efficient but realistic estimation of the core damage frequency, and
- provide an estimate of the frequency of each IE or IE group.

Preparing a Preliminary List of IEs for the ACR-700

In its report *Systematic Review of Plant Design for Identification of Initiating Events* [5], AECL provided a detailed review of its process for identifying IEs for the ACR-700 using master logic diagrams (MLDs). The focus of these MLDs was to identify all initiators that could lead to any unintended release of radioactive material. The MLDs were then extended to identify systems/components required to keep the radioactive material in place or to keep it from being released to the public, along with possible failures of these systems or components that would result in releases to the public. AECL's analysis identified 200 potential initiators. AECL then removed 34 IEs estimated to have very low frequencies or consequences. An additional four IEs were judged to be flooding initiators rather than reactor transient initiators and were also screened from the evaluation. The remaining 162 events were then grouped by similarity of plant response or bounding consequences, resulting in 87 grouped events.

The scope of *Preliminary Design Assist PSA Level 1—Selected Full Power Event Trees* [6] was to develop and analyze event trees for selected IEs that ". . . are judged to produce high values of 'individual-sequence' SCDF [severe core damage frequency] and/or be major contributors to 'summed' SCDF . . ." in the ACR-700. Consequently, this document provides information regarding IEs that are judged by AECL to be of importance to the risk profile of the ACR-700. All 11 IEs were within the list of 87 grouped events identified by AECL through the use of MLDs.

Supplementing the ACR-700 List of IEs With an Assessment of CANDU Operating Experience

Additional information utilizing the experience of currently operating CANDU-designed reactors aided in performing a critical evaluation of the ACR-700 IEs. The *Generic CANDU Probabilistic Safety Assessment* [7] provides an analysis of the CANDU 6 design that AECL considers to be representative of the existing CANDU 6 plants in operation or under construction at that time. Like the ACR-700 analysis, this systematic evaluation first prepared MLDs to identify the sources of radioactive material and supplemented it with a systems/component analysis. A total of 76 IEs were identified for the CANDU 6. Because of differences in plant designs, not all of the CANDU 6 IEs were applicable to the ACR-700, and vice-versa.

The IEs that were screened out or left out in the ACR-700 and CANDU 6 documentation were reviewed with individual decisions made about whether to return the screened (or omitted) IE to the final list. Each preliminary ACR-700 IE was then assigned a frequency based on the ACR-700, CANDU 6, or U.S. NPP information using conservative criteria, and a final set of preliminary IEs for the ACR-700 was identified. Thus, a comprehensive list of IEs with estimated frequency values were independently derived from ACR-700 and CANDU 6 information. In comparing these 87 IEs for the ACR-700 to the CANDU 6 list of 76 IEs, 5 of the 34 IEs that were screened from consideration because of low frequency were returned to the preliminary list of IEs for future consideration.

Comparing the Preliminary ACR-700 List of IEs to U.S. Industry Experience

The ACR-700 and CANDU 6 information was critically examined and compared to U.S. NPP experience to reflect the expectations for an ACR-700 plant being licensed and operated within the United States. A comprehensive list of possible IEs at a U.S. light-water moderated pressurized-water reactor (PWR) was derived from NRC and nuclear industry lists. The lists from historical documents (WASH 1400 [8], EPRI-NP-2230 [9], and NUREG/CR-3862 [10]), NUREG/CR-5750 [11], and two advanced reactors (AP600 [12] and CE80+ [13]) were collated into one table (including IE frequencies). Because the methods used by AECL to identify IEs were judged to be consistent with U.S. practices, a consistent technical basis existed for comparing the U.S. NPP IE experience and the ACR-700 IE information. This compilation identified several sequences that appear applicable to the ACR-700 but were not listed (e.g., the ACR-700 listed the single closure of an MSIV but not the simultaneous closure of all MSIVs) and identified 16 IE frequencies that differed by over an order-of-magnitude from U.S. operating experience (e.g., the partial loss of 120 V dc was 3 orders-of-magnitude lower than the frequency reported in U.S. IPEs).

Over the years, regulators, designers, and safety analysts have noted that a small number of unique plant scenarios can present a significant potential for core damage and releases to the environment. These special conditions are usually treated as IEs in the design of NPPs in the United States. The conditions include station blackout (SBO), anticipated transients without scram (ATWS), and interfacing system loss-of-coolant accidents (ISLOCAs). These IEs are not part of AECL's 87 grouped events and were added to the final list of independently derived IEs for the ACR-700. In addition, for all CANDU reactors, including the ACR-700, systems exist that permit refueling of the reactor during operation. Because this "at-power" activity is not possible with U.S. NPP designs, there is no U.S. experience for such activities. ORNL developed an MLD to identify the radiological barriers for the fuel handling system. Based on this, several transient and radiological releases associated with on-line refueling were recommended such that the scope of the IEs more closely matched those of NPPs licensed in the United States.

Systems Review

An important element in the understanding of a PRA involves gaining knowledge of the various plant systems that would be depended upon to prevent or mitigate IEs that could reasonably be postulated to occur and that, if unmitigated, could lead to fuel damage and/or radionuclide releases. To develop the plant system models, a list of those systems that provide or support safety functions was required. However, more than 200 systems and structures were identified in the reviewed ACR-700 design documentation [14, 15], out of which more than 100 were designated by AECL as "safety-related." The purpose of this review was to focus on identifying and reviewing those systems that are (1) the most important (per AECL) in mitigating the risk-significant IEs and their consequences and/or (2) similar to the systems in U.S. NPPs where risk-significant events have been identified and analyzed.

The process of selecting a list of ACR-700 systems with an impact on plant behavior in accident scenarios and applicable to other advanced reactor designs was conducted in two steps. First, applicable CANDU 3/6 and ACR-700 documentation was reviewed, and the resulting information was then supplemented with and compared to U.S. LWR experience.

Systems Important to Safety According to CANDU 3/6 and ACR-700 Documentation

Because not all systems important to safety are the same for each generation of CANDU plant, these differences were reviewed to ensure that a system important to safety was not overlooked in the information compilation process. The following documentation was used in the system selection process:

1. *Preliminary Design Assist PSA Level 1—Selected Full Power Event Trees* [6],
2. *System Analysis of the CANDU 3 Reactor* [16], and
3. *Generic CANDU Probabilistic Safety Assessment—Reference Analysis* [7].

A review of the 11 event trees that model those IEs deemed by AECL to be the most risk significant [6] (based on CANDU operating experience) identified 12 groups of systems used to mitigate the occurrence of those IEs. A similar review of the systems listed as important to the CANDU 3 plants [6] identified two additional systems for consideration that were not identified in the list of IEs compiled for the ACR-700 event trees─the shutdown cooling system (SDCS) and the moderator liquid poison system (MLPS). In the CANDU 3/6 design, the SCDS provides for the removal of decay heat during normal reactor shutdown. In the ACR-700 design, this system was eliminated and instead, its function was included as part of the long-term cooling (LTC) portion of the emergency core cooling system (ECCS). The CANDU 3 report also credits the MLPS as a means to shutdown the reactor. In the ACR-700, the MLPS has the safety function to "maintain the reactor in a shutdown condition" and apparently has no mitigating function while the reactor is at power. Thus, the MLPS was not retained on the list of systems important to safety.

A review of generic CANDU 6 event trees highlighted two supplementary systems not captured in ACR-700 list─the emergency water supply (EWS) system and the moderator cover gas system. The EWS system does not exist in the ACR-700 design, but part of its functionality is recognized in the reserve water system (RWS). In both the CANDU 6 and the ACR-700 designs, the moderator cover gas system circulates the helium gas that is present above the moderator surface in the calandria vessel. The loss of the cover gas in the calandria vessel can lead to the release of radionuclides into the containment. This system appears to be a very specific system that comes into play only in the moderator-related IEs; therefore, it was not included in the initial list of ACR-700 systems to be analyzed.

Of the systems identified in the three reviews, only the RWS is a new system in the ACR-700 design that did not exist in the previous CANDU designs. The RWS "delivers emergency water by gravity into various process systems to provide and/or facilitate an essentially passive, interim heat sink" [6]. Thus, the list of systems important to safety as derived from a compilation of systems used to mitigate IEs is the same as the 12 groups of systems identified by AECL for the ACR-700.

Incorporating U.S. Operating Experience into the List of Systems Important to Safety

The 12 grouped systems discussed above were then compared to and supplemented with U.S. LWR experience. First, the general design criteria (GDC) found in Appendix A of 10 CFR 50 [17] were reviewed to identify specific systems that are explicitly required for the plant design.[*] These include containment, electric power, reactor protection, reactivity control, reactor coolant makeup, residual heat removal, emergency core cooling, containment heat removal, containment atmosphere cleanup, and cooling water. With the exception of the containment system, all of these systems had already been included on the ACR-700 systems list compiled in the AECL documentation review. The containment system was added to the list of systems to review.

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[*] The GDC establish the minimum requirements for the principal design criteria for light-water-cooled NPPs in the United States. The GDC "are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units." It is important to note that Appendix A further states that "additional or different criteria will be needed to take into account . . . water-cooled nuclear power units of advanced design."

Next, Chapter 15 of NRC Regulatory Guide (RG) 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants* [18] was reviewed to identify systems that are designated as safety-related or are used to mitigate the consequences of an accident. RG 1.70 identifies 28 representative initiating events (IEs) grouped into 8 categories for analysis. The systems or components compiled from the eight categories of events include feedwater, main steam, reactor coolant system, onsite and offsite ac power, pump shaft, ECCS, pressurizer safety relief valves, chemical and volume control (CVCS), steam generator tubes, condenser, and turbine. All of these systems had been reviewed in the previous steps of the ACR-700 system selection process, with the exception of the CVCS. The function of the CVCS appears to be performed by the pressure and inventory control system in the ACR-700 design. Therefore, this review based on information provided in RG 1.70 did not identify any new systems requiring review for the ACR-700.

Finally, NUREG-1560 [19] was examined to determine if there are systems that are important to safety based on the IPE data for U.S. NPPs that may be similarly important for the ACR-700. NUREG-1560 is a summary report prepared by the NRC on 75 IPE submittals pertaining to 108 U.S. NPPs. Most of the systems highlighted in the review of NUREG-1560 have an equivalent system in the ACR-700 design that had previously been included on the systems list with two exceptions—the IAS and the HVAC systems. The IAS was added to the list of systems to be studied in detail. However, although the HVAC systems are extremely important to safety in U.S. NPPs, the inclusion of the HVAC systems onto the list of systems was placed on hold until further information on the system design became available from AECL.

Review of Systems Important to Safety

The review of U.S. operating experience and licensing criteria resulted in the addition of two systems to the original list of 12 systems and also provided verification of the ACR-700 system list derived from other sources. For each of the 14 systems, a brief summary of the system function, a description of major components, and a discussion of the interfaces with other systems as derived from reviewed ACR-700 documentation were developed. Qualitative insights from this information along with comparisons with U.S. LWR designs were also provided. Finally, ACR-700 information gaps of importance to this information compilation effort were called out.

There are significant differences between the LWR designs that have been licensed for operation in the United States and the ACR-700 design. The use of heavy water moderation, refueling of the core during power operation, and the use of 292 individual core flow channels present design features that simply do not exist in U.S. LWR designs. These design differences will require further review to determine their actual risk significance and whether the associated systems reflect reliability commensurate with their risk significance.

Also several ACR-700 systems seem to mirror U.S. LWR systems and applications. These systems include the instrument air, main steam, main and auxiliary feedwater, seal cooling, and service water systems. However, the information reviewed indicates that although the overall function of the ACR-700 system may be similar, there seem to be some specific differences between the ACR-700 and the U.S. LWR designs. A review of more detailed design information is needed before these design differences can be fully understood and their impacts evaluated.

In addition, several ACR-700 systems have system functions comparable to U.S. LWR designs, but the available information indicates the ACR-700 has significant design differences with regard to implementing those functions. These systems include reactor shutdown, heat transport, electric power, emergency core cooling, primary pressure and inventory control, and containment. The systems have roles in performing primary safety functions of reactor shutdown, maintaining primary inventory integrity, and mitigating accident consequences; all are functions that have been shown from U.S. risk studies of NPPs to be significant contributors to risk if compromised. Therefore, these design differences will need to be more thoroughly examined as further information becomes available from AECL.

Finally, there are systems that reflect the features of advanced LWR designs more than they reflect existing designs. For example, reserve water is provided by gravity to various systems when needed. As more specific design information becomes available from AECL, these features will need to be reviewed to ensure that they satisfy current licensing requirements.

PRA Review Tools Selection

The selection of appropriate PRA review tools to be used in assessing the preliminary ACR-700 design included regenerating the tools and models that were used in the evaluation of the CANDU 3 design and documented in NUREG/CR-6065, *Systems Analysis of the CANDU 3 Reactor* [16]. The regeneration task was performed on current computational platforms, and the results were documented and compared to the results found in NUREG/CR-6065.

Systems Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE) was selected as the computer software program used for the analysis. The fault trees and event trees documented in NUREG/CR-6065 were duplicated, and the model's output cut sets were determined to be identical to those documented in NUREG/CR-6065.

PROTOTYPE RISK EVALUATION MODEL DEVELOPMENT

A prototype (i.e., proof-of-concept) risk model for the ACR-700 NPP was developed using SAPHIRE. One event tree (small LOCA) and its supporting fault trees were modeled with the supporting basic event data developed from the ACR-700 systems information. In constructing the model, the standard U.S. LWR PRA approach to a small LOCA was followed, while still reflecting the ACR-700 design. The "proof-of-concept" study successfully demonstrated that the model worked and could be further developed, thereby providing an independent risk evaluation model to assess the ACR-700 design and supporting risk analyses.

CONCLUSIONS

In preparation of a possible design certification review of the ACR-700, the NRC began examining selected areas of nuclear safety, identifying accidents that could potentially dominate the risk profile of the ACR-700 design, and reviewing or evaluating other risk-important design and technology issues. To support the use of PRA in all regulatory matters, descriptive information was prepared on the IEs and systems necessary to evaluate the plant's risk profile. In addition, a prototype of a model that would be capable of evaluating an independently developed PRA was constructed. The compilation and use of relevant U.S. operating experience aided in developing a comprehensive list of IEs and also helped in evaluating what events are likely to be considered should the NRC receive an application to license the ACR-700 for operation in the United States.

While this work has been concluded due to a change in industry direction, the documentation developed under the project provides a concrete basis and clear direction for further analysis efforts to confirm the risk profile of the ACR-700 in accordance with NRC licensing requirements.

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NOTICE

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