THE ADVANCED NEUTRON SOURCE SAFETY APPROACH AND PLANS

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Abstract

The Advanced Neutron Source (ANS) is a user facility proposed for construction at the Oak Ridge National Laboratory for all areas of neutron research. The neutron source is planned to be a 350-MW research reactor. The reactor, currently in conceptual design, will belong to the United States Department of Energy (USDOE). The safety approach and planned elements of the safety program for the ANS are described. The safety approach is to incorporate USDOE requirements [which, by reference, include appropriate requirements from the United States Nuclear Regulatory Commission (USNRC) and other national and state regulatory agencies] into the design, and to utilize probabilistic risk assessment (PRA) techniques during design to achieve extremely low probability of severe core damage. The PRA has already begun and will continue throughout the design and construction of the reactor. Computer analyses will be conducted for a complete spectrum of accidental events, from anticipated events to very infrequent occurrences.

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1. INTRODUCTION

The Advanced Neutron Source (ANS) will be a new user facility for all kinds of neutron research, including neutron scattering, materials testing, materials analysis, isotope production, and nuclear physics experiments (see companion paper entitled "The Advanced Neutron Source: Designing to Meet the Needs of the User Community," by F. J. Peretz). The thermal neutron source is a compact heavy-water cooled and reflected reactor nominally rated at a fission power of 350 MW. The project is currently in the conceptual design stage, and operation is planned for the year 1999. Key design parameters are given in Table 1. The aluminum-clad cermet fuel to be used for the reactor is based on the concept successfully utilized at Oak Ridge National Laboratory during a 20-year period of operation of the 100-MW High Flux Isotope Reactor (HFIR). The ANS cermet will combine aluminum powder and the U_3Si_2 fuel material developed at Argonne National Laboratory; HFIR fuel continues to be manufactured with U_3O_8.

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Table 1. Specifications and typical design parameters for the Advanced Neutron Source reactor

Quantity	Nominal value	Notes
Fission power level, MW(f)	350	
Power transferred to primary coolant, MW(c)	332	Heat convected away from fuel plates
Average power density, MW(c)/L	4.9	
Maximum power density, MW(c)/L	8.3	Estimated, fuel grading not yet optimized
Core life, d	14	
Core active volume, L	67.4	Fueled volume
Fuel form	U_3Si_2	
Fuel matrix	Al	
Volume of fuel in meat, %	15	
Fuel loading, kg ^{235}U	14.9	
Fuel cladding	6061 Al	
Fuel plate thickness, mm	1.27	
Clad thickness, mm	0.254	
Coolant channel gap, mm	1.27	
Coolant (and reflector)	$D_2O(D_2O)$	
Inlet pressure, MPa	3.7	
Inlet temperature, °C	49	
Heated length, mm	474	
Coolant velocity in core, m/s	27.4	May be reduced after detailed analysis
Core pressure drop, MPa	1.6	
Outlet pressure, MPa	2.1	
Bulk coolant outlet temperature, °C	81	
Average heat flux, MW(c)/m^2	6.3	
Maximum heat flux, MW(c)/m^2	10.7	Estimated, fuel grading not yet optimized
Maximum fuel centerline temperature, °C	400	Design limit
Peak thermal flux in reflector, 10^{19} $m^{-2} \cdot s^{-1}$	>8	Unperturbed

The ANS reactor design retains many of the inherent safety advantages of research reactors. For example, the primary coolant exits from the core at a temperature well below 100°C even at full power, so a loss-of-coolant accident (LOCA) places no special loads upon containment because the subcooled primary coolant cannot flash to steam. The reactor pool surrounds the reactor and much of the primary coolant piping with a massive heat sink that would be very useful for a variety of plant emergencies and would act to retain fission products in the event of hypothetical severe accidents. A pipe break accident under water would result in depressurization and a very small loss of primary coolant, but would not, because of the surrounding pool water, lead to uncovering the core.

The small core mass of the ANS limits the amount of energy that could be released in postulated severe accidents and also limits the amount of hydrogen or deuterium gas that could be produced via oxidation of the metallic clad in the event of melting. Frequent refueling minimizes the buildup of long-lived fission products and provides ample opportunity for equipment testing and maintenance. The use of heavy water (D_2O) coolant and reflector result in a relatively long neutron lifetime, which makes the reactor power level respond relatively slowly to reactivity upsets and enables the reactor protection system to stop power excursions before fuel damage even for rapid insertion of prompt critical quantities of reactivity.

The ANS has one safety feature that has not always been included in the design of research reactors: a leak-tight containment building. The primary purpose of the containment building is to retain radioactive fission products in the event of severe accidents for which complete core destruction is postulated. This feature minimizes or eliminates the dependence upon evacuation of residents from areas adjoining the U.S. Department of Energy (USDOE) reservation for severe accident emergency planning (the option of evacuation will, of course, be preserved in emergency planning).

Along with the many inherent safety advantages, design features that warrant additional consideration must be identified and studied, and compensatory safety measures taken if needed. The average power density in the active fuel region is 4.9 MW/L--high, but necessarily so, if the ANS is to fulfill its mission of providing a very high flux of neutrons for beam research. A satisfactory level of safety will result from the planned course of fuel behavior research and development, probabilistic risk assessment (PRA)-based facility design, transient and steady-state safety analyses, and, finally, by a stringent quality assurance and inspection program for the fuel manufacture. Adequate thermal-hydraulic margin for normal operation and anticipated transients is to be ensured by including appropriate uncertainty factors in the design and safety analyses.

Part of the normal thermal-hydraulic margin is provided by primary coolant pressurization (~3.7 MPa at core inlet), so special attention must be devoted to the transient thermal-hydraulic analysis of depressurization accidents. In operation and design, stringent measures will be taken to eliminate the possibility for pipe leakage to progress to pipe rupture.

The many thin, closely spaced fuel plates provide a large surface area for transfer of the thermal energy from the fuel, but the close spacing introduces a vulnerability to channel flow blockage. This vulnerability is greatly reduced by the use of a full-flow, fine mesh (smaller than the 1.27-mm spacing between fuel plates) screen in the main primary coolant circulation path. In addition, refueling equipment and procedures are designed to eliminate the introduction of foreign material into the primary coolant system. Loose parts detection and flow blockage monitoring will be a part of the precritical and power escalation process followed after every refueling.

2. SAFETY GOALS AND OBJECTIVES

A comprehensive safety program is in place to ensure that the ANS reactor design has a level of safety commensurate with modern standards. The ANS will belong to the USDOE and will be subject to its regulations that require compliance with standards at least as strict as those of the U.S. Nuclear Regulatory Commission (USNRC). The reactor is, therefore, being designed to meet applicable USNRC regulations and standards; safety analysis reports, with a format and content compatible with the USNRC requirements, will be produced in phase with the project construction and operation schedules.

Recent USDOE policy developments in nuclear safety philosophy have stressed safety awareness and accountability, defense-in-depth, and probabilistic risk assessment. A recent draft of the USDOE Nuclear Safety Policy Statement provides qualitative safety goals for individual and societal radiological risks that are very similar to those prescribed by the familiar USNRC reactor safety goals policy[1] and states two very stringent quantitative guidelines for the risk associated with new USDOE reactors: (1) the probability of severe core damage or meltdown at individual new USDOE reactors should normally be less than one per one hundred thousand reactor years, and (2) the frequency of accidents accompanied by severe releases of radioactivity should normally be less than one in a million reactor-years.

PRA is a basic part of the ANS safety program. PRA studies were initiated at the preconceptual stage[2] and will continue through to facility operation. The PRA will function not only to demonstrate the level of facility safety, but also to guide the design effort. The degree of sophistication of the analyses and the scope of interaction with the design will change and evolve as the facility design matures. Very early in the project, interaction between the PRA analysts and reactor designers lead to the decision to place the primary coolant system largely under water to minimize the consequences of a pipe break accident. More recently, PRA studies have shown that the "leak-before-break" approach accepted by the USNRC[3] can be used to greatly reduce the probability of occurrence of a pipe rupture accident.

3. THE USE OF PRA IN DESIGN

The interaction between the PRA and the design efforts is currently taking place in several different modes, each of which is directed at determining what design features will be necessary to meet the USDOE guideline for severe core damage probability of one per hundred thousand years. For example, the project has recently adopted more ambitious design criteria for seismic acceleration than previously envisioned. A peak ground acceleration of 0.5 g, consistent with a return period of between 10,000 and 100,000 years for the Oak Ridge area, has recently been adopted as a design goal for the reactor and related systems. The USDOE guideline for severe core damage is also being used to allocate the reliability requirements for cooling and safety system reliability requirements.

The event tree for a loss of off-site power (LOSP) accident will be used to illustrate the process of determining failure probability goals for each major plant system (or function), so the reactor as a whole can meet the USDOE guideline for severe core damage probability. Figure 1 displays the LOSP event tree; only branches that lead to core damage are shown. The probability of the initiating LOSP may be conservatively set at one per year based on experience in the Oak Ridge area; the various subsequent branch failure probabilities are to be allocated by examination of the LOSP and other event trees.

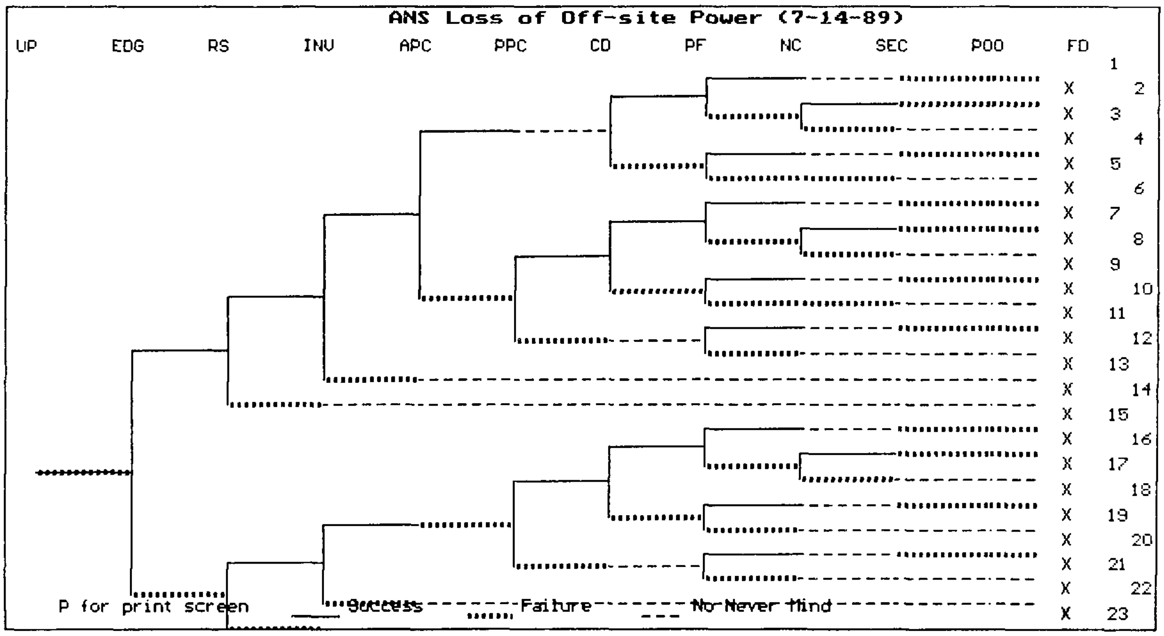

Fig. 1. Event tree for loss of off-site power.

LEGEND

UP = utility power
RS = reactor scram
APC = active pressure control, primary coolant
CD = primary coolant pump extended coast-down
NC = natural convection primary coolant flow, post-scram
POO = reactor pool post-scram heat sink
EDG = emergency diesel generator
INV = inventory retention function, primary coolant
PPC = passive pressure control, primary coolant
PF = forced primary coolant flow, post-scram
SEC = secondary coolant system, post-scram heat sink
FD = fuel damage

Determination of failure probability goals for individual plant systems is done under the ground rule that the core damage probability for each branch of the event tree should be less that one per million reactor years, because the total of the core damage probabilities of all the branches should not exceed one per hundred thousand. This ground rule leads immediately to several obvious conclusions, particularly for the branches for which core damage results from the failure of only one system function. For example, referring to branch 14 on Fig. 1, the reactor shutdown function failure probability must not exceed one per million demands (i.e., following LOSP) if the fuel damage probability for branch 14 is not to exceed one per million. For this reason the ANS reactor protection system consists of two independent shutdown systems, one of which utilizes control rods inside the reactor core pressure boundary tube (CPBT) and another set of rods outside the CPBT. Either of these two shutdown systems can reliably and independently shutdown the reactor after a LOSP.

Other functional failure probability goals are being set in a similar manner, although the process becomes more complex for branches that require multiple failures for core damage. In such cases, the failure probabilities must be allocated by examining the safety and operability benefits to be gained by different combinations of system failure limits. After overall sub-system failure probability goals are set, the next step will be to guide the system design effort by utilizing fault trees to determine which different subsystem component configurations can best achieve the selected goals.

Another strategy for minimizing risk in the design and operation of the ANS is to utilize the lessons learned from similar reactors. In this regard, the PRA of the HFIR facility at Oak Ridge is being studied intensively. An effort is currently under way to adapt and utilize the insights gained from the HFIR PRA. The adaptation effort accounts for differences between the HFIR design and the conceptual ANS design to ascertain which events present the dominant possibilities for severe core damage. Preliminary results indicate that primary coolant flow blockage is the dominant severe core damage initiator for the ANS, just as it is for the HFIR. For both reactors, the probability of severe core damage is held to an acceptable level by the use of full-flow, fine mesh filtration and by stringent material control procedures during refueling. Therefore, even the dominant severe accidents are not expected to occur over the life of the facility. The HFIR has never experienced even local fuel damage from any cause during a 20-year period of operation.

Both a Level I PRA (core damage probabilities) and a conservative Level II PRA (containment release source term probabilities) will be produced for the ANS preconstruction safety assessment, and the Level II PRA will be extended to include off-site consequences before the completion of construction.

4. ACCIDENT ANALYSIS

A broad spectrum of accidents has been postulated and will be analyzed for design purposes and for the ANS safety analysis report. The acceptance criteria for accident analysis are selected to match the calculated probability of occurrence of the accident with the allowable consequences. The more frequently an accident is expected to occur, the milder must be the consequences. For anticipated events (those of estimated frequency of occurrence of > 0.01/year) the fuel must remain within acceptable design limits. At the other end of the spectrum, minor fuel damage would be acceptable for accident categories having probability of occurrence between 1 in 10,000 per year and 1 in 1 million per year. Table 2 lists the accident categories and acceptance criteria and provides examples of the events in each category. Most of the accident categories are the familiar types of upsets involving mismatches between core power level and core heat removal, but some are unique to research reactors or to the ANS design, such as events involving the beam tubes or the cryogenic cold sources.

The RELAP-5 computer code[4] has been selected as the primary tool for transient thermal-hydraulic analysis. Developed at Idaho National Engineering Laboratory (INEL), RELAP-5 has become an accepted tool for the analysis of LOCAs for pressurized-water reactors. RELAP-5 has recently been modified to include heavy water properties and to improve numerical convergence in the low-pressure region. Additional changes are required to improve the applicability of the thermal-hydraulic correlations in RELAP-5 to the narrow channel, parallel plate ANS fuel geometry. This includes the correlations that describe the heat transfer or heat transfer limitations,[5] and those that describe two-phase flow phenomena. A package of code modifications has been completed that includes the following: the Petukhov correlation[6] for single-phase turbulent wall-to-fluid heat transfer, a critical heat flux correlation developed specifically for the ANS,[7] utilization of a formerly available interfacial heat transfer model to represent properly the vapor generation rate during subcooled boiling, and the modification of the interfacial drag terms in the slug flow regime to duplicate the void-quality relationship predicted using the drift-flux model as developed by Griffith.[8]

The RELAP-5 code will be validated for application to the ANS by utilizing operational transients recorded at other USDOE research or test reactors. Separate effects experimental data will provide additional validation of thermal-hydraulic correlations in the code. Small-scale steady state and transient experiment loop tests are planned at ORNL to test the validity and robustness of the thermal-hydraulic correlations, particularly those for critical heat flux and net vapor generation.

A severe accident methods development program has been initiated at ORNL to study the severe accident issues as they relate to high power density research reactors and to produce the needed analytical tools for severe accident analysis. The ANS design goal limiting severe fuel damage probability to less than one per hundred thousand years will make the occurrence of a severe accident at the ANS a very hypothetical consideration, but there are two practical reasons for including severe accident studies in the safety analysis program. The first is the commitment of the USDOE to the concept of defense-in-depth. This requires that the facility be designed to contain the radionuclides that might be released from the fuel by a severe accident, even if the occurrence of severe fuel damage is extremely unlikely. The ability to design for the containment of severe accidents requires an understanding of the physical and chemical phenomena associated with severe accidents. The other reason for including a severe accident study in the safety program is to support the PRA effort. The assessment of risk requires quantification of consequences because risk is the product of the probability of occurrence and the consequences of occurrence.

Table 2. ANS design basis events and acceptance criteria

Event class[a]	Estimated frequency (per year)	Unrestricted area radiation exposure goals	Fuel cooling, temperature conditions
Normal e.g., Startup Power adjustments Shutdown	≥ 1	0.005 rem/year total body (10 CFR 50, Appendix I)[b,c]	No boiling at hot spot, fuel temperature below long-term limit
Anticipated e.g., Loss of off-site power Uncontrolled single control rod withdrawal Small coolant leaks Loss of cold source Single pump failure Single valve failure Pressure control malfunction Loss of reflector coolant flow	< 1 > 1/100	0.025 rem/year effective dose equivalent (40 CFR 61, Subpart H)[d]	Critical heat flux not exceeded at hot spot, fuel temperature below short-term design limit
Unlikely e.g., Uncontrolled all-rod withdrawal Coolant flow screen blockage (partial) Medium coolant leaks Secondary coolant pipe break Cold source pressure boundary fault Extended loss of off-site power	< 1/100 > 1/10000	0.5 rem/year effective dose equivalent (10 CFR 20)[e]	No fuel melting

Table 2 (continued)

Event class[a]	Probability (per year)	Unrestricted area radiation exposure goals	Fuel cooling, temperature conditions
Extremely unlikely e.g., Primary coolant flow screen blockage (major) Major primary coolant pipe rupture Cold source internal explosion	< 1/10000 > 1/1000000	25 rem/event effective dose equivalent (10 CFR 100)[f] 1 to 5/event guideline for emergency planning	No wide-spread fuel damage
Beyond design basis	< 1/1000000	25 rem/event effective dose equivalent; 1 to 5/event guideline for emergency planning	Not applicable

[a]Event groupings are based on current approximations of probabilities and are subject to change.

[b]10 CFR refers to Title 10, ("Energy") of the *U.S. Code of Federal Regulations.*

[c]10 CFR 50 ("Domestic Licensing of Production and Utilization Facilities"), Appendix I provides numerical guidelines for design objectives and limiting conditions for operation to meet the criterion "as low as is reasonable achievable" for radioactive material in light-water-cooled nuclear power reactor effluent.

[d]40 CFR 61, ("Protection of Environment") Subpart H ("National Emissions Standard for Radionuclide Emissions from Department of Energy Facilities") specifies annual limitations for radiation exposure to any member of the public in the vicinity of a USDOE reservation.

[e]10 CFR 20 ("Standards for Protection against Radiation") provides upper limits (in the context of the "as low as is reasonably achievable" doctrine) for radiation exposures and radioactivity concentrations in liquid and airborne effluent in restricted and unrestricted areas.

[f]10 CFR 100 ("Reactor Site Criteria") specifies limiting off-site radiation doses for hypothetical severe reactor accidents.

The immediate objective of the severe accident task is to perform scoping studies to identify the severe accident issues, possible design implications, and needed modifications to existing severe accident analysis computer codes. The severe accident methods development task has been initiated early in the design process to allow severe accident considerations to have an impact upon the development of the design.

5. CONCLUSIONS

The ANS approach to safety is to meet or exceed both USDOE regulations and policies and applicable USNRC requirements and to maximize the degree of safety of the facility design. A comprehensive safety program is in place; it relies heavily upon PRA techniques, but also devotes significant resources to the understanding of the physical phenomena of accidents and to the development of computational tools for predicting the consequences of a wide spectrum of accidents. The safety program has been initiated very early in the design process to allow every opportunity for safety analysis results to affect the facility design and to ensure that the completed reactor facility meets the highest standards for nuclear safety.

6. REFERENCES

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