Advanced Neutron Source (ANS) Project
Progress Report
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ACRONYMS

1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional
ABB	Asea Brown Boveri
ACAR	angular correlation of annihilation radiation
ACRS	Advisory Committee on Reactor Safeguards
AECL	Atomic Energy of Canada, Limited
ALARA	as low as reasonably achievable
ALWR	advanced light water reactor
AMBK	ANS materials databook
AMIS	ANS materials information system
ANL	Argonne National Laboratory
ANS	Advanced Neutron Source
ANSDM	ANS Dynamic Model
ANSR	ANS reactor
ASIC	application-specific integrated circuit
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox Company
BNL	Brookhaven National Laboratory
BSR	Bulk Shielding Reactor
CAD	computer-aided design
CALS	Continuous Acquisition and Life-Cycle Support
CANDU	Canadian deuterium uranium reactor
CBCF	carbon bonded carbon fiber
CCB	change control board
CDR	conceptual design report
CEBAF	continuous electron beam accelerator facility
CECE	combined electrolysis and catalytic exchange
CETF	control element test facility
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CHF	critical heat flux
CPBT	core pressure boundary tube
CRADA	cooperative R&D agreement
CRBRP	Clinch River Breeder Reactor Plant
CSAR	conceptual safety analysis report
DCH	direct containment heating
DEGB	double-ended guillotine break
DIO	digital input/output
DOE	U.S. Department of Energy

DORT	discrete ordinates transport code
DTTA	delta-T/Tavg
EPRI	Electric Power Research Institute
EPROM	erasable, programmable, read-only memory
FBTF	flow blockage test facility
FCI	fuel coolant interaction
FE	flow excursion
FFTF	fast flux test facility
FSAR	final safety analysis report
GUI	graphical user interface
HFIR	High Flux Isotope Reactor
HWUDF	heavy water upgrade and detritiation facility
IB	incipient boiling
IFD	inertial flow diode
ILL	Institut Laue-Langevin
INEL	Idaho National Engineering Laboratory
ISOL	isotope separation on-line
JAERI	Japan Atomic Energy Research Institute
LDV	laser doppler velocimetry system
LEU	low enriched uranium
LOSP	loss of off-site power
M&C	Metals and Ceramics (Division)
MCCI	molten core-concrete interaction
MCNP	Monte Carlo neutron photon
MCP	main circulation pump
MMES	Martin Marietta Energy Systems, Inc.
MOI	maintenance and operation integration
MORRZ	maintenance, operation, research, and radiation zoning
NISA	Numerically Integrated Elements for System Analysis
NIST	National Institute of Standards and Technology
NP-HWR	New Production-Heavy Water Reactor
NRC	U.S. Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
NUREG	Nuclear Regulatory Guide
Nu	Nusselt
OCR	outer control rod
ORNL	Oak Ridge National Laboratory
ORO	Oak Ridge Operations
ORR	Oak Ridge Reactor
OSV	onset of significant voiding
PCO	project change order
PDR	plant design requirements
Pe	Peclet
PIN	project information network
PRA	probabilistic risk assessment
PSAR	preliminary safety analysis report
PSF	positron spectroscopy facility
PSVA	primary supply vessel adapter

PSVAW	primary system vessel adapter weld
PWR	pressurized-water reactor
R&D	research and development
RAM	reliability, availability, and maintainability
RCS	reactor coolant system
RDA	rapid depressurization accident
RERTR	Reduced Enrichment for Research and Test Reactors
RO	reactor operator
ROAAM	risk-oriented-accident-analysis-methodology
RPCS	reactor primary coolant system
RPS	reactor protection system
RSCS	reactor secondary cooling system
RTD	resistance temperature detector
RV	reflector vessel
RVCS	reflector vessel cooling system
S&Z	Saha and Zuber
SAP	severe accident program
SAW	severe accident workshop
SDD	system design description
SPND	self-powered neutron detectors
SRI	Southern Research Institute
St	Stanton
T/H	thermal-hydraulic
THTL	T/H test loop
TLC	thermoluminescent crystals
TPRL	Thermophysical Properties Research Laboratory
TREAT	transient reactor test
W&F	Whittle and Forgan
WBS	work breakdown structure

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FOREWORD

The President's budget request for FY 1994 included a construction project for the Advanced Neutron Source (ANS). However, the budget that emerged from the Congress did not, and so activities during this reporting period were limited to continued research and development and to advanced conceptual design.

A significant effort was devoted to a study, requested by the U.S. Department of Energy (DOE) and led by Brookhaven National Laboratory, of the performance and cost impacts of reducing the uranium fuel enrichment below the baseline design value of 93%. The study also considered alternative core designs that might mitigate those impacts. The ANS Project proposed a modified core design, with three fuel elements instead of two, that would allow operation with only 50% enriched uranium and use existing fuel technology. The performance penalty would be 15–20% loss of thermal neutron flux; the flux would still just meet the minimum design requirement set by the user community. At the time of this writing, DOE has not established an enrichment level for ANS, but two advisory committees have recommended adopting the new core design, provided the minimum flux requirements are still met.

Colin D. West
ANS Project Director

PROJECT MANAGEMENT

1

Although the Advanced Neutron Source (ANS) did not gain line-item status in FY 1994, the project team continued to put in place the management and control procedures that would be required in a construction line item. A graded, but rigorous, change control process was in place and was used.

The funding level during this period was lower than in the previous year, when staffing levels had been built up in anticipation of the capital funding for construction that was included in the President's budget request. Accordingly, the number of staff working on the project was steadily reduced, with appropriate attention to priority areas and to new work, such as the enrichment study initiated by the U.S. Department of Energy (DOE). These changes were carried out successfully while meeting budget constraints and milestones.

The ANS configuration management system was explained in the last progress report. Since that time, experience has been gained through the processing of nine additional project change orders (PCOs) for Change Control Board (CCB)-1, the highest level of change. Currently, two PCOs are being processed for CCB-1 and two for CCB-2. This experience has been used to improve and simplify the configuration management process. The configuration management plan has been updated twice since its initial publication as part of the conceptual design report (CDR). These updates dealt with many items that were "to-be-determined" in the initial issue; added material to define the process

further; more clearly defined "change-types" and CCB responsibilities; and more clearly defined responsibilities of the project and administrative staff in the application of the process. The associated project procedures, which implement the configuration process, have also been updated to remain consistent with the configuration management plan. Finally, a data base is being used to track each PCO through the process, to collect the action items resulting from each PCO, and to monitor the accomplishment of these action items.

One more revision in the configuration management plan and its associated procedures is underway in preparation for Title I design. This revision will further define the relationship between the configuration management system and the management approval and sign-off process. It is anticipated that this extensive experience in configuration control before the arrival of the industry team will result in a smooth transfer of responsibilities once the full team is in place.

1.1 ANS PROJECT QUALITY PROGRAM

In concert with the reduced funding of the project during FY 1994, the ANS quality group had less to support and reduced its activities accordingly; total support to the project in this area was less than three-quarters of a man-year.

The principal activity of the ANS quality group was to review and revise the *Advanced*

Neutron Source Quality Assurance Plan to meet the requirements of DOE rule 10 (CFR) Code of Federal Regulations 830.120, which became effective midyear. Although not many provisions of the ANS quality program changed, the development of a detailed implementation plan to serve as the compliance basis of the program presented the greatest challenge, particularly since DOE had not identified the format and content desired for an implementation plan. A workable format has now evolved throughout the DOE complex and is being used to provide a common style for such plans.

The ANS Project led a quality audit of the Babcock and Wilcox Company (B&W) Research

and Test Reactor Fuel Elements Group, which is performing much of the ANS fuels development activity. The team was composed of members of the DOE research reactors complex whose fuel is fabricated by that organization. The audit verified the implementation of the quality program and satisfactory completion of quality improvement actions identified during the previous audit.

The ANS quality group's general support to the project was centered around the limited ongoing activities in research and development (R&D), definition of reactor fuel and core configuration, and proposed site characterization. This support consisted mainly of participation in drawing, procedure, and readiness reviews.

RESEARCH AND DEVELOPMENT

2

R&D activities in thirteen areas have been identified as essential to the ANS Project. These R&D activities are needed to address feasibility issues, to produce the data necessary to make a rational decision when alternative design concepts are identified, and to examine and demonstrate the applicability of technological advances. This chapter summarizes progress on these activities for the reporting period and includes tasks performed by Argonne National Laboratory (ANL), B&W, Battelle Pacific Northwest Laboratory, Elemental Research Application Software, Idaho National Engineering Laboratory (INEL), Martin Marietta Energy Systems, Inc. (MMES), Central Engineering Services, the Massachusetts Institute of Technology, the National Institute of Standards and Technology (NIST), Oak Ridge Associated Universities, Oak Ridge National Laboratory (ORNL), Tennessee Technological University, Southern Research Institute (SRI), the University of Rhode Island, the University of Tennessee, and the University of Virginia.

2.1 REACTOR CORE DEVELOPMENT

2.1.1 Core Design

Much of the core design effort in this period was devoted to support of DOE's study of the impacts of reducing the enrichment of

the ANS fuel. The study, led by Brookhaven National Laboratory (BNL), also included consideration of modifications to the core design to incorporate low enriched fuel (Bari et al. 1993). The core configurations selected for study were the baseline, two-element design (Copeland et al. 1989, West 1989); a larger volume (82.6-L) optimized three-element design that had been developed as a possible alternative earlier in the project (Copeland et al. 1989); and a hypothetical, still larger (108-L) three-element design selected for neutronic studies, although it was recognized that the dimensions were not mechanically practical or safe. A further description of the optimized cores is found in Sect. 3.2. Further work, beyond the BNL study, led to the establishment of correlations among enrichment, fuel properties, core life, and neutron flux (Alston, Gehin, and West 1994).

The fuel used in all high performance research reactors in the Western world is a mixture of fine aluminum powder, to provide a good thermal conductivity, and particles of a uranium compound. The compound chosen for ANS is U_3Si_2 , which was developed and tested worldwide through the fuel development work of the Reduced Enrichment for Research and Test Reactors (RERTR) Program.

The enrichment of the fuel may be lowered by adding increasing proportions of the nonfissile ^{238}U isotope into highly enriched ^{235}U . Of course, this change in proportions reduces the amount of the useful isotope in a given volume or mass of fuel; therefore, to maintain the mass

of ^{235}U necessary to sustain the chain reaction and to produce excess neutrons for beams and other experiments, the volume devoted to the uranium, the density of uranium in the fuel, or both, must be increased as the enrichment is reduced. The feasibility of increasing the fuel density was considered by an independent group of experts in the BNL study and, later, by a DOE review team; their findings are summarized in Table 2.1.

The minimum acceptable core life, in order to meet the users' requirements for availability and predictability, is 17 d. This core life results, for any given core design and fuel density, in a "cliff" in the relationship between thermal neutron flux and enrichment. As enrichment is

reduced, the flux at first falls slowly (primarily, resulting from absorption of neutrons by the nonfissile ^{238}U), but when the enrichment is reduced to the point where full power cannot be maintained for 17 d, the operating power must be reduced to maintain the minimum 17-d core life, and the neutron flux falls, almost in proportion to the power reduction. This phenomenon is illustrated in Fig. 2.1.

The "enrichment cliff" can be postponed to lower enrichments by using a higher density fuel, but, for the baseline design, a successful new fuel development would be needed to reduce the enrichment below about 70% without passing the cliff and incurring very large performance losses

Table 2.1. Fuel development: uranium density, technical risk, and reactor safety

Uranium density (g/mL)	BNL ^a study expert evaluation	DOE ^b review of ANS ^c fuel development program	Notes
>6.0	<10% probability of success in development program		
>4.8			ANS Project calculations show that the low thermal conductivity of such fuels would necessitate a reduced reactor power for safe operation
6.0–3.5	50–95% probability of success in development programs		BNL panel also found that "going to fuel loadings of 4.8 g U/cm ³ and higher will introduce larger costs and uncertainties and require considerable development effort"
<3.5		Four (of five) panel members believe this fuel could be qualified with minimal cost and schedule impact	
3.5–1.3	95–100% probability of success in development programs		
<2.2		The fifth DOE panel member believes that this fuel could be qualified with minimal cost and schedule impact	
1.7			Baseline ANS design

^aBNL = Brookhaven National Laboratory.

^bDOE = U.S. Department of Energy.

^cANS = Advanced Neutron Source.

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Fig. 2.1. Performance of baseline core design: 330-MW maximum power, 2.2 g/mL fuel, 17-d core life.

(Fig. 2.2). However, the baseline core design is very highly optimized for highly enriched (93% ^{235}U) fuel, and there is little hope of developing fuels to give a high neutron flux with this design at an enrichment much below about 50% (Fig. 2.2).

The studies reveal that an optimized core composed of three elements, which is compared with the baseline two-element configuration in Fig. 2.3, would provide enough extra volume to operate with lower enrichments with practical fuel densities (Fig. 2.4). However, the larger core volume brings a 15–20% penalty in the neutron flux produced, although the unperturbed thermal neutron flux at the beam tube mouths would still be (just) greater than five times that in the Institut Laue-Langevin (ILL) reactor. The extra volume greatly reduces the fuel density needed for high performance operation at the lower enrichments.

It should be noted that the larger volume and shorter heated length of the modified core design may lead to still larger thermal-hydraulic (T/H) safety margins than the baseline core.

The costs of a three-element core would be considerably higher than the baseline (Bari et al. 1993), but this can be mitigated or even avoided if advantage is taken of the larger fuel inventory to increase the core life and, therefore, reduce the number of fresh cores needed each year. These studies continue: at the time of writing it appears

Fig. 2.2. Various fuel densities.

Fig. 2.3. Core designs.

likely that DOE will adopt the three-element design, but no decision on enrichment level has been made yet.

2.1.2 Thermal-Hydraulics

A significant amount of T/H analysis has been performed to help develop the conceptual

design and to establish operating margins and parameters. These efforts have included thermal and hydraulic analyses of the fuel elements as well as surrounding components; additional uncertainty analyses, including data base development and correlation selection; development of T/H models to predict two-phase pressure drop within the fuel element coolant channels; and thermal analysis of fuel-plate

Fig 2.4. The modified core design can take better advantage of higher density fuels if they are successfully developed. (Example shown is a 17-d life core with 330-MW maximum power level.)

manufacturing defects, as well as others. Some of these efforts are briefly described here.

2.1.2.1 Thermal-Hydraulic Analysis

A series of parametric calculations was performed to evaluate approximate operating parameters for an 82.6-L, three-element core design. This evaluation was accomplished by using existing two-element power distributions to generate maximum power limits for the three-element design. Figure 2.5 shows an example of the calculations performed. The limiting thermal power levels for various T/H criteria are presented as a function of coolant velocity (all at a 95% nonexceedance probability level). Error bars on the figure represent potential variations in the power distributions for the actual three-element design, pending detailed calculations. These results indicate that the three-element design should be able to operate at 303 MW(t) [330 MW(f)] with an inlet pressure of 3.2 MPa without exceeding the oxide or incipient boiling (IB) limits that are presently design criteria.

Hydraulic analysis was expanded to evaluate the hydraulic forces acting on the inner control rods and was used to develop more detailed cooling designs that prevent an inner control rod from being ejected if it was sheared from the latch and drive mechanism. In addition, the analysis identified the orificing required to cool the lower portions of the rod and the designs that minimize lift forces. The analysis included buoyancy, viscous, and pressure forces acting on the rods. Parametric evaluation of coolant velocities within the control rod region was used to establish the maximum coolant velocity that can be used without risk of ejecting even a sheared-off rod.

2.1.2.1.1 Uncertainty Analysis

Uncertainty analysis has included incorporation of additional data in the T/H data base. A total of 338 flow excursion (FE) data points from 13 different sources has been added to the existing data base. These data points

Fig. 2.5. Limiting core powers for the three-element fuel design as a function of coolant velocity for an inlet pressure of 3.2 MPa.

extend the data base to velocities as low as 0.1 m/s and subcooling levels as low as 0.75°C. These data have been used to support changing from the Costa (1967) onset of significant voiding (OSV) correlation, which is presently being used by the project, to a modification of the Saha and Zuber (S&Z; 1974) correlation (discussed in Sect. 2.4).

Preliminary statistical analyses of U_3Si_2 fuel plate nonhomogeneity defects have now been performed on 12 plates manufactured with the High Flux Isotope Reactor (HFIR) outer fuel element contour. The fueled portion of the plates was divided into axial and radial regions for these analyses in order to provide meaningful information about heat transfer to coolant channels along the plates. The results quantify for silicide plates what was already known for HFIR plates with U_3O_8 fuel—fuel loading tends to be somewhat skewed toward the

center of the plates, with overloading occurring there and underloading near the edges. Loading variations are also larger near the edges of the plates, with the largest variation appearing on the edge of the plate. The results are very good, and local deviations were all significantly lower (i.e., better) than the present specification of a maximum of 20% overload over a 2-mm-diam spot.

T/H analyses using “worst-case” uncertainty levels were performed in order to evaluate the feasibility of meeting very low core-melt probabilities. For these analyses, parameter uncertainties were all assumed to occur simultaneously in an unfavorable manner at two to three standard deviation levels. This approach was employed in HFIR safety analysis and is considered to be very conservative. Results of these preliminary calculations indicate that low exceedance probability levels appear to be

achievable but that the ANS design and the treatment of uncertainties must be optimized to reach ANS design goals and requirements.

2.2 FUEL DEVELOPMENT

Work continued in support of the U_3Si_2 -Al dispersion as the reference fuel for ANS, and the irradiation behavior of backup candidate fuels was studied. Work in the areas of fabrication development, irradiation performance, and modeling are discussed below.

2.2.1 Fabrication Development

Fabrication development at B&W during the year concentrated on demonstrating the feasibility of achieving dual fuel gradients. All the methods investigated involved varying the thickness of the constant volume fraction fuel/aluminum dispersion along the width and length of the fuel plate. The approach taken was to impose a longitudinal taper on the bottom of a compact that has a width gradient on the top produced by the proven method used for HFIR fuel plates.

Best results were obtained when the lengthwise gradient was maintained during rolling by a solid metal filler. (The widthwise gradient was maintained by a compacted powder filler similar to HFIR plates.) The spotwise homogeneity was improved by eliminating the coarse fuel particles and by limiting the maximum fuel particle size to 200 mesh ($74\ \mu m$). This production method produced double grading plates with good reproducibility, at least over the small number of plates produced so far. The plates were essentially identical; however, these limited studies did not involve varying fuel or aluminum powder lots. The local variations in the 2-mm-diam spot loading were only about $\pm 5\%$. Work in the next year will include determining the feasibility of placing the thin portion of the fuel meat in the center of the plate thickness, further studies of reproducibility with different powder lots, and fabrication of higher density fuel loadings.

Work was begun at ANL to evaluate the fabricability of fuel plates made using atomized (spherical) U_3Si_2 powder. Atomized powder potentially can be produced much more economically than the comminuted powder being used today. This work will be scaled up to full-sized plates at B&W.

2.2.2 Irradiation of Fuel Capsules

Additional postirradiation examinations of HANS-1 silicide samples have been completed, and the results have not altered the U_3Si_2 fuel behavior description and model presented in the FY 1993 annual report.

The main postirradiation analysis effort this year was concentrated on the HANS-2 samples. This second HFIR irradiation test contained samples of backup candidate fuels UAl_x , UAl_2 , and U_3O_8 , as well as additional U_3Si_2 samples. The irradiation conditions of HANS-2 were similar to those of HANS-1, and metallographic examination of the U_3Si_2 samples revealed, indeed, no difference in irradiation behavior of the U_3Si_2 . Therefore, no further examinations were deemed necessary for those samples.

Metallography of the aluminide and oxide samples showed significant temperature effects on their high-burnup (~ 90 atom % ^{235}U) irradiation behavior. The appearance of UAl_x (see Fig. 2.6) irradiated at $250^\circ C$ is identical to that observed in numerous previous irradiation experiments. It is noteworthy that, even at a burnup of 90% of the ^{235}U atoms in highly enriched UAl_x , no fission gas bubbles can be found. The behavior is quite different at $425^\circ C$, however, as shown in Fig. 2.7. Extensive interlinked porosity has formed in the aluminide particles, the composition of which, because of high uranium burnup, is now somewhere between UAl_4 and aluminum. Evidently, the fission gas mobility in this composite material at 250 and $425^\circ C$ is sufficiently different that bubble coarsening occurs at the higher temperature even though it does not at the lower.

Fig. 2.6. Micrograph of UAl_x fuel particle irradiated at 250°C . Composition of the irradiated particle is 80AL-7U+FP . (500X, as polished.)

Fig. 2.7. Micrograph of UAl_x fuel particle irradiated at 425°C . (500X, as polished.)

The situation for UAl_2 is, in principle, similar, but porosity at 425°C is much less extensive, presumably because the higher initial uranium content leaves the fuel particles with a much higher UAl_4 fraction after 90% burnup than in UAl_x .

As expected, the U_3O_8 particles had completely reacted with the aluminum matrix at both 250 and 425°C . The fuel behavior at 250°C appeared to be similar to that in previous irradiations (i.e., extensive fission gas porosity had formed in the reaction phases). The most

surprising finding in the HANS-2 test was the microstructure of the reacted U_3O_8 at 425°C (as shown in Fig. 2.8) no fission gas porosity is evident in either the light (metallic) or dark (oxide) phase. It may be concluded that both backup candidate fuels are prone to fission gas bubble formation at high burnup (aluminide at high temperatures and oxide at low temperatures) even though each appears to be remarkably stable at the opposite end of the ANS fuel temperature range. Because the ANS fuel must be stable over the entire temperature range and because the relatively low uranium density of these compounds requires high fuel volume fractions in the meat, translinkage of fission gas bubbles throughout the meat is a distinct possibility with either of the backup candidate fuels.

Fig. 2.8. Micrograph of U_3O_8 fuel particle after irradiation at 425°C . Compositions of the phases in the irradiated particle are $5\text{Al-50U-20RE-10MoZr+FP}$ (light) and 70Al-5U-15Si+FP (dark). (500X, as polished.)

The HANS-3 capsule has been completed and inserted in the HFIR target basket and will be irradiated in the HFIR cycle beginning in early October 1994. The HANS-3 fuel specimens consist of punchings from rolled plates and are thus more prototypic than the pressed aluminum-fuel mixtures irradiated in the first two capsules. The specimens consist primarily of U_3Si_2 bracketing the range of expected ANS

temperatures. There are some aluminide and oxide backup fuels along with some lower enrichment U_3Si_2 specimens (40 and 20% ^{235}U) to study the effects of lower burnup and fission rate.

2.2.3 Fuel Performance Studies and Modeling

High energy heavy ion irradiations and low dose neutron irradiations at well controlled temperatures have continued in an effort to understand the basic irradiation processes involved in fuel behavior. While fission during the neutron irradiations is the same as during fuel burnup, the ion irradiations produce damage similar to end-of-fission track damage. The advantage of ion simulations is the speed with which they can be performed, the ability to vary experimental parameters, and their lack of radioactivity, which permits access to a wider range of experimental facilities. The two techniques complement each other, and their results are in agreement.

Ion irradiations have identified the temperature ranges for amorphization and breakaway swelling in U_3Si and for amorphization in U_3Si_2 as well as a microstructural transformation at higher temperatures that provides resistance to amorphization. Current work includes neutron irradiations of U_3Si and U_3Si_2 at 30 and 350°C monitored by neutron diffraction and ion simulation of aluminum mixing into and diffusion in U_3Si , U_3Si_2 , USi , $USi_{2-\beta}$ and USi_2 .

The neutron irradiations are aimed at understanding the fundamental mechanisms responsible for amorphization and the high temperature microstructural transformation of U_3Si and U_3Si_2 . The hope is to find chemical additions that produce the same stable structures. The neutron irradiations at 350°C indicate a change in defect configuration that will be investigated by ion irradiation and electron microscopy. Ion simulation of aluminum mixing into and diffusion in the uranium silicide systems indicate changes of radiation-enhanced diffusion with uranium content and temperature that may

alter steady-state defect fluxes and swelling during burnup.

The fuel performance model, DART (Rest and Hofman 1991), has continued to evolve in order to explain the observed irradiation behavior. DART-C thermal conductivity and bubble distribution calculations were performed for various U_3Si_2 and UAl_x fuel loadings in support of ORNL hot spot analyses. DART-C combines the analysis of the behavior of dispersion fuel plates (previously based on a single-phase model of fuel swelling) with a new analysis for the description of multiphase fuel swelling.

A theoretical model for irradiation-induced amorphization and crystallization was developed in order to facilitate the resolution of U_3Si_2 fuel behavior issues between ion-bombardment studies and observations of in-reactor fuel performance. This new model complements the existing model of irradiation induced grain subdivision (recrystallization) and provides an interpretation of the ion-irradiation experiments on normal and preirradiated U_3Si .

Recent transmission electron microscopy work by the Japanese (Nogita and Une 1994) on high burnup UO_2 fuel has essentially verified the physical picture that forms the basis of the theory of irradiation-induced grain subdivision and swelling in U_3Si_2 and UO_2 developed by Rest and Hofman (1991), and generalized with the underlying mechanisms further elucidated in a recent publication (Rest and Hofman 1994). The results of the new model of irradiation-induced amorphization and crystallization are qualitatively consistent with the nanocrystalline interpretation of the neutron diffraction results on the low enriched uranium (LEU) U_3Si_2 aluminum dispersion fuel.

2.3 CORROSION TESTS AND ANALYSES

The main objective of the corrosion test program is to ensure that film growth and corrosion processes on the aluminum alloy fuel cladding of the ANS core do not lead to excessive temperatures or otherwise compromise the containment capabilities of the clad. To that

end, experiments continue in the ex-reactor test facility under a wide range of T/H conditions typical of those anticipated for ANS operation. The results of this ongoing test program have been reported in detail in the previous ANS progress reports (Selby, Harrington, and Peretz 1990, 1991, 1992, 1993, 1994), ORNL technical reports (Pawel et al. 1990a, 1993a), and open-literature publications (Pawel et al. 1990b, 1991; Yoder et al. 1992; Pawel, Felde, and McFee 1993).

The accumulated test information has resulted in the building of a data base and data correlation that permit the prediction of the extent of corrosion product growth as a function of several important system parameters. Such a predictive capability is an essential part of the iterative design process that considers both the T/H and the neutronic aspects of the reactor operation.

2.3.1 Summary of Results: FY 1994 Update

Test equipment problems associated with the main loop circulation pump and the test section power supply were essentially resolved by the start of this fiscal year, and eight more tests have been carried out since. The emphasis this year has been in the examination of the corrosion behavior of "alternate alloys" for comparison with the reference 6061 aluminum cladding material upon which the present data correlations are based. While reproducibility issues, some apparently associated with the sensitivity of the film growth process to small variations in the coolant water chemistry, have precluded completely unequivocal quantitative comparisons, useful information on film growth and, particularly, the characteristics of spallation were found.

Experiments have been conducted with alternate cladding alloys 8001 aluminum, 1100 aluminum, and (in progress) a special French alloy used in the fuel cladding of the ILL reactor. Specimens of pure aluminum were also subjected to corrosion loop tests for comparison. The alternate alloy tests were all

accomplished under nearly identical T/H and water chemistry conditions at a high average heat flux, 12.5–14 MW/m². The film growth kinetics, while showing some departures from the norm in both the shape of the rate curves and the calculated rate constants, were not considered significantly different from the behavior exhibited in the 6061 aluminum data base. Most of these variations were "conservative" in that they generally involved growth rates that decreased more than anticipated for the long-time exposures in the loop.

A significant difference in the alternate alloy corrosion behavior relating to film spallation and its consequences was found. In previous tests with 6061 aluminum, spallation of the oxide was not observed for steady-state conditions until the temperature drop across the oxide film, ΔT_{ox} , had exceeded 120°C (Pawel et al. 1990a, 1991). The majority of the calculated values of ΔT_{ox} at spallation was in the range of 130–140°C. Moreover, in the alloy beneath the spalled oxide, an internal reaction zone rapidly formed. The nature of this zone was such that it would influence both heat conduction and the mechanical properties of the cladding. In the present effort, both the initiation and subsequent impact of film spallation were found to be dependent upon the particular alloy. The pure aluminum specimen exhibited an extremely early spallation (relative to our criterion) at a ΔT_{ox} of only about 100°C. In addition, the internal reaction zone was observed to be very extensive, aggressively deforming the metal and even causing some expansion into the flow channel.

On the other hand, the 8001 aluminum and the ILL alloy were much more resistant both to the onset of spallation and to the subsequent internal reaction buildup. Two experiments were conducted with 8001 aluminum: the first was stopped when the maximum ΔT_{ox} was greater than 170°C, but no spallation or internal reactions were observed; the second was run for an extended period, increasing the average heat flux in the latter stages from 12.5 to 14 MW/m², finally producing spallation at the hot end of the specimen at $\Delta T_{ox} \simeq 180^\circ\text{C}$. Metallographic examination of this specimen indicated that no

internal reactions had occurred, even though spallation had been progressing for almost 100 h before the experiment was terminated. Although the initial indications are that ILL alloy behaves similarly to the 8001 aluminum, detailed evaluations of this test have not been completed.

The tests this year confirmed that the oxide film growth is dependent on coolant water chemistry, and that a particularly sensitive region is near pH 5, obtained by nitric acid additions to deionized water, where most of the experiments have been conducted. An example of this influence is given in Fig. 2.9, which is a plot of the "rate factors" [i.e., the ratio of the measured rate constant for boehmite ($\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$) film growth to that predicted by data Correlation

II; Selby, Harrington, and Peretz 1993; Pawel, Felde, and McFee 1993] for a specific set of experiments as a function of the coolant pH. The selected data are from tests with coolant inlet temperatures, T_{ci} , less than 50°C and reasonably aggressive T/H conditions where oxide-coolant interface temperatures exceed 120°C . The scatter in the data near pH 5 underscores the large difference in growth rate that exists over a single unit variation in the pH. It is important to point out that the two solid circles in the figure were obtained in a single test where the effective coolant pH was suddenly changed from 5.05 to 4.8, with all the other test parameters remaining the same. A rather arbitrary curve has been drawn through the data points to emphasize

Fig. 2.9. The effect of coolant pH on the relative oxide film growth rate constants for certain aluminum alloy specimens exposed in the Advanced Neutron Source corrosion test loop. (Note: The two solid circles in the figures were obtained in a single test where the effective coolant pH was suddenly changed from 5.05 to 4.8.)

the concept of a dramatic change in rate in the vicinity of pH 5. A similar plot can be made using the more limited data from tests in which the inlet coolant temperature was in the 75–80°C range.

Additional tests in the pH 4.5–4.9 region are planned to provide a better definition of the behavior at pH levels below 5.0 and to serve as a basis for recommendations for coolant water chemistry control in the ANS reactor (ANSR).

2.4 CORE FLOW TESTS

The T/H performance of the ANSR will directly influence both its normal and off-normal operational limits and behavior. Experiments are being performed to allow the development and validation of models used in the T/H analysis. The experimental plan is designed to allow examination of phenomena that are critical to characterizing the T/H behavior of ANS and that must ultimately be incorporated in applicable computer codes and models. The plan also allows, under some circumstances, for the collection of data to support statistical analysis of the reactor performance. Thermal limit tests performed in the T/H test loop (THTL) and flow blockage tests performed in the flow blockage test facility (FBTF) along with supporting analytical modeling are described below.

2.4.1 Thermal-Hydraulic Loop Tests

The THTL facility was designed and built to provide known T/H conditions for a simulated full-length coolant subchannel of the ANSR core, thus facilitating experimental determination of FE and critical heat flux (CHF) thermal limits under expected operational conditions. Detailed descriptions of the test facility and the test section design are given in Felde et al. (1994) and in last year's ANS annual report (Selby, Harrington, and Peretz 1994).

The THTL experimentation was focused initially on FE and CHF phenomena (Siman-Tov

et al. 1993). Since then, additional experiments have been performed that expand the data range to much higher heat fluxes (18 MW/m²) and velocities (28.4 m/s).

A summary of the data in the form of plots of pressure drop vs mass flux for both the FE and CHF tests is shown in Fig. 2.10. The heat fluxes indicated for each test correspond to the channel average values at the critical point (minimum pressure drop or actual burnout) of that particular test.

Acquiring FE data at these high heat fluxes and velocities is significant for two reasons. First, very few data are available for FE at velocities higher than 10 m/s. In the initial part of this activity, the range was extended to 21 m/s and has now been extended further to 28.4 m/s (at the exit), which is above the maximum velocity expected in the ANSR, including uncertainties. Second, the high heat flux achieved in this study is well above both the ANSR nominal average heat flux of 5.9 MW/m² and the nominal peak heat flux of 12 MW/m², and is as high as the "hot channel peaking factor" heat flux (e.g., peak heat flux with all uncertainties) of 18 MW/m². The limiting velocity corresponding to the ANSR nominal peak of 12 MW/m² is about 20 m/s (at a subcooling of 23°C), well below the ANSR nominal velocity of 25 m/s.

Preliminary results from tests with a test section of wider span (25.4 mm) indicate no major effect on critical velocity as a result of the increased span dimension. Final conclusions on the subject, however, await additional experiments with larger span test sections.

The collected data were compared with correlations by Costa (1967), Whittle and Forgan (W&F; 1967), and S&Z (1974). In the Costa correlation, the FE heat flux is proportional to the square root of velocity, whereas both W&F and S&Z (as well as most other FE correlations) show a linear dependence. The S&Z correlation shows the best agreement with the data, whereas the Costa correlation becomes increasingly conservative with increasing mass flux. These and other considerations support a shift from the Costa correlation to the S&Z correlation

Fig. 2.10. Flow excursion data from thermal-hydraulic test loop experiments.

for the ANSR, especially since, for most of the ANSR scenarios analyzed (including nominal conditions), the mass fluxes are quite high. Plotting both the THTL data and the rest of the ANS data base for FE (Siman-Tov et al. 1991) in terms of the Stanton (St) number [the selected dimensionless group for Peclet number (Pe) > 70,000 in the original S&Z correlation] against subcooling reveals a clear trend for St [and, to a lesser degree, Nusselt (Nu) number] to increase with decreasing subcooling, a trend that becomes stronger at lower subcoolings. In addition, both St and Nu must by definition become infinite for zero subcooling, which contradicts constant values for these parameters as suggested by the original S&Z correlation. This observation, in combination with the trend observed in the data, provides a strong case for modifying the

S&Z constant St (0.0065) and Nu (455) criteria, especially in the low subcooling range. Based on the best fit with the ANS data base, the following modifications to the original S&Z correlations were developed and proposed for ANSR applications on a preliminary basis:

$$St_{FE} = 0.0065 \left(0.55 + \frac{11.21}{dT_{sub}} \right),$$

$$Pe_{FE} > 70,000; \quad (1)$$

and

$$Nu_{FE} = 455 \left(0.55 + \frac{11.21}{dT_{sub}} \right),$$

$$Pe_{FE} < 70,000; \quad (2)$$

where

dT_{sub} = subcooling temperature difference =
 $T_s - T_b$ (K),

Nu_{FE} = Nusselt number at the FE point,

Pe_{FE} = Peclet number at the FE point,

St_{FE} = Stanton number at the FE point,

T_b = bulk coolant temperature (K),

T_s = bulk coolant saturation temperature (K).

Figure 2.11 shows the selected modified correlations in terms of St and Nu , respectively, against the ANS data base, including the THTL data of the present study. Statistical evaluation of these modified S&Z correlations shows considerable improvement in the means and the standard deviations in comparison with those of the original S&Z and Costa correlations. Additional investigation of these proposed

modifications is still in progress in order to determine their final acceptable form. A more detailed discussion of the experiments and results and the proposed modifications and their constraints is given in Siman-Tov et al. (1994).

2.4.2 Flow Blockage Tests

The flow blockage issue was studied with the goal of minimizing the impact of any possible blockage scenario on reactor safety. This year's activities were focused on characterizing the fluid flow patterns within the channel as a function of blockage size and location. Given this characterization, several design parameters can be modified to enable safe reactor operation in the presence of a potential flow blockage. These parameters include the length of the

Fig. 2.11. Comparison between the modified and unmodified forms of the Saha and Zuber correlation.

unheated entrance region or the provision of entrance screens or alternate cross-channel flow paths.

The flow patterns within a blocked channel were evaluated using three parallel avenues. First, experiments were conducted to measure the wall heat transfer coefficients over a limited portion of the channel wall near the channel entrance. Second, experiments were conducted to measure the fluid velocity vectors and turbulence characteristics at six discrete slices across the cooling channel. Third, two computational fluid dynamics (CFD) codes were used to model the flow patterns in both blocked and unblocked channels.

All experiments were performed at the FBTF. The FBTF provides a full-scale, full-velocity, well-instrumented replica of a single core cooling channel with the ability to vary inlet blockage size and position, as shown in

Fig. 2.12. A low-power heater plate was included to provide a measurable temperature difference between the fluid and the channel wall over a 30-mm long section of the wall near the channel entrance. Thermoluminescent crystals (TLC) were applied to the back of the thin heater plate, and the resulting color patterns were recorded with a video camera. Calibration techniques were developed to permit the translation of the color images to wall temperatures and heat transfer coefficients. The results showed that the region of reduced heat transfer extended beyond the 30-mm test section for all tested blockage sizes.

A new laser doppler velocimetry system (LDV) was installed to provide direct measurement of the fluid velocity vectors across the channel span for six axial locations, ranging from a point 27-mm downstream from the blockage to one 187-mm downstream. Tests were made with both central and edge blockages at

Fig. 2.12. Schematic diagram of the flow blockage test facility.

both full and reduced channel pressure drops (the reduced pressure drop tests were used to provide additional benchmark data for the CFD models). Backflow recirculation regions behind edge blockages were clearly revealed by these experimental measurements. However, for centrally positioned blockages, the LDV system measured no net backflow, instead recording regions of highly turbulent, reduced-velocity forward flow.

Modeling efforts using the FLOW3-D CFD code, (Fluent 1993) based on a transient solution algorithm, were not successful. This code required extremely long computational times and was unable to predict the unblocked flow velocity

corresponding to the design pressure drop accurately. It was, however, useful in evaluating the effect of simplifying assumptions necessary to model the plenum region of the FBTF.

The Fluent CFD code (Fluent, Inc., 1993) provided much quicker turnaround, enabling a parametric study of blockage size and position effects to be made. Benchmark comparisons to experimental data show that the CFD results for both unblocked channels and channels with edge blockages are very good, as shown in Fig. 2.13 and in Table 2.2. In these cases, the flow pattern is well matched, and the heat transfer coefficients within the affected region are conservatively underestimated by Fluent. For central blockages

Fig. 2.13. Comparison of experimental laser doppler velocimeter data and Fluent computational fluid dynamics results for a 25% edge blockage in the flow blockage test facility.

Table 2.2. Experimental and analytical results from the flow blockage evaluation

Condition	Reattachment length (mm)	Axial ^a location for h and V (mm)	Convective heat transfer coefficient, h (kW/m ² · K)	Average channel fluid velocity, V (m/s)
Unblocked channel, 1.44-MPa pressure/drop, water at 305 K				
Fluent	N/A ^b	34	100	24.3
Dittus Boelter ^c	N/A	34	133	25
TLC ^d	N/A	34	105	25.1
Unblocked channel, 1.44-MPa pressure drop, D₂O with variable properties for average 6 MW/m² wall heat flux				
Fluent	N/A	50	130	25.4
Dittus Boelter	N/A	50	110	25
Fluent	N/A	500	150	25.6
Dittus Boelter	N/A	500	140	25
25% edge blockage, 1.44-MPa pressure drop, water at 305 K				
Fluent	65	27	20 ^e	23.6
TLC	N/A	27	45 ^e	24.3
DB/LDV ^f	58 < X < 90	27	18 ^e	26.1
35% center blockage, 1.44-MPa pressure drop, water at 305 K				
Fluent	57	27	47 ^g	23.3
TLC	N/A	27	104 ^g	23
DB/LDV	X < 27	27	51 ^g	25

^aMeasured from cooling channel inlet.

^bN/A = not applicable.

^cBased on design velocity.

^dTLC = thermoluminescent crystals.

^eMeasured 10 mm from channel edge, directly behind blockage.

^fFrom Dittus Boelter correlation, corresponding to velocity measured by laser doppler velocimeter (LDV) with a 1.45-MPa channel pressure drop

^gMeasured at channel center.

however, Fluent predicts a strong recirculation zone behind the blockage that is not shown by either the TLC or LDV data. An investigation

of both experimental and computer model factors is underway to understand this remaining discrepancy.

2.5 CONTROL CONCEPTS

As part of the reactor kinetics activity, the prompt-neutron lifetime, an important kinetics parameter needed for safety analysis, was reevaluated this past year. The analysis was carried out using a two-dimensional (2-D) VENTURE (Vondy, Fowler, and Cunningham 1990) model with smeared reflector components. The reflector components must be included in the model to account accurately for the decrease in neutron transport back to fuel as a result of the neutron scattering and absorption in the components. As a point of reference, the measured prompt neutron lifetime for ILL ranged from 0.80–0.94 ms without the boron burnable poison and beam tubes (depending upon control rod position) down to 0.61–0.68 ms with the boron burnable poison and beam tubes. The ILL-calculated values, using methods similar to the ANS approach, were within approximately 2% of the measured values.

The calculated prompt-neutron lifetime for ANS using this modeling approach when reflector components were included in the model was 0.54–0.63 ms, depending upon the control rod position. The corresponding range for the case without reflector components was 1.35–1.5 ms. The higher prompt neutron lifetime of ANS compared to ILL for the noncomponent case is most likely a result of a larger heavy water reflector in ANS, and the larger relative change in prompt neutron lifetime between the noncomponent and the component case for ANS is most likely the result of a larger total reactivity effect from reflector components in ANS.

Calculation of the prompt-neutron lifetime using the ANS Monte Carlo neutron photon (MCNP; Breismeister 1986) model was not possible because MCNP calculates a nonimportance-weighted value that is generally longer than the prompt-neutron lifetime. For example, using the ANS MCNP model without reflector components, and, without importance weighting, a lifetime of about 5 ms is calculated. The lifetime must be importance-weighted to account properly for the neutrons slowing down in the reflector. The current recommendation is

to use a conservative value of 0.5 ms with the expectation that the value could increase slightly.

2.6 CRITICAL EXPERIMENTS

Various facilities are under consideration for performing the ANS critical experiments. However, in 1993, DOE instructed the project to place the activity on hold pending the results of the low enrichment study. Although there was hope throughout FY 1994 that a resolution of this issue could be obtained, no final decision was made, and, thus, there was no progress in this activity during the report period. It is hoped that DOE will make some decision before the middle of FY 1995 that will allow resumption of this important task leading to the selection of a site for these experiments by the end of the next fiscal year. It is believed that, regardless of the fuel enrichment, this activity needs to move forward to avoid becoming a critical path issue.

2.7 MATERIAL DATA, STRUCTURAL TESTS AND ANALYSES

2.7.1 Code Case for 6061-T6 Aluminum

Early in the project, and after careful consideration of candidate materials, 6061-T6 aluminum was selected as the reference core pressure boundary tube (CPBT) structural material. Although used in previous research reactors, 6061-T6 aluminum was not at that time included in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code for Class 1 nuclear construction. The ANS Project formally requested the ASME B&PV Code Committee to include 6061-T6 aluminum as an acceptable material for Class 1 components in 1990. *Code Case N-519—Use of 6061-T6 and 6016-T651 Aluminum for Class 1 Nuclear Components, Section III, Division 1* has now been approved by the ASME B&PV Code Committee and published in Supplement

2.7.2 Fracture Toughness of Irradiated 6061-T6 Aluminum

The extremely high neutron flux in the ANSR is expected to cause significant embrittlement eventually as aluminum atoms are transmuted to silicon by thermal neutrons. Extensive irradiation testing and surveillance programs are planned to determine the irradiation embrittlement of 6061-T6 aluminum more precisely. The first capsule assembly, HANSAL-T1, was irradiated at 95°C in HFIR to a peak thermal neutron fluence of 1.1×10^{26}

m^{-2} ($E < 0.5 \text{ eV}$). Eight of the 16 compact specimens in the second capsule HFIR assembly, HANSAL-T2 that were irradiated at 95°C at a fluence of $8 \times 10^{26} \text{m}^{-2}$, were tested during this report period. The results from both capsules are shown in Fig. 2.14. Preliminary examination of the data indicates that fracture toughness at 25 and 95°C is not greatly reduced by irradiation equivalent to that which the ANS CPBT will receive in 6 months of operation. Although the fracture toughness at 150°C was reduced, the reduction was less than expected, the minimum value of toughness being $22 \text{ MPa}\sqrt{\text{m}}$. These preliminary data offer encouragement that the CPBT can operate safely for its minimum design life of 6 months, and perhaps longer.

Fig. 2.14. Fracture toughness as a function of test temperature for 6061-T651 in the unirradiated condition; after irradiation to 10^{26}m^{-2} at 95°C (HANSAL-T1 experiment); and after irradiation to $8 \times 10^{26} \text{m}^{-2}$ at 95°C (HANSAL-T2 experiment).

Weld specimens were irradiated in a third capsule assembly, HANSAL-T3, for comparison with the base metal material previously irradiated in HANSAL-T1 and -T2. The specimens from HANSAL-T3 and the remaining specimens from HANSAL-T2 will be tested in FY 1995.

2.7.3 Appraisal of Radiation Effects on Structural Components in ANS

An in-depth assessment of the suitability of 6061 aluminum alloy to withstand the specific service conditions of the ANS environment is under way. A comprehensive survey of radiation effects on the mechanical and physical properties of 6061 alloy has been completed. It defines the information gaps pertinent to ANS, namely effects of irradiation temperatures greater than 100°C, high stresses, and very strongly thermalized neutron spectra. Available mechanical properties data consist primarily of tensile data and the fracture toughness data discussed above. Measures to fill the gap and to identify ways of ensuring adequate performance of the ANS components have been initiated.

The first objective is to minimize irradiation temperatures and operating stresses to avoid the mechanical degradation that is characteristic of the 6061-T6 alloy under long-term exposure to them. The CPBT will be required to sustain a persistent hoop stress that may, perhaps, cause accelerated creep under irradiation, resulting in permanent barreling of the CPBT. The extent of such deformation cannot be estimated yet because there are no radiation creep data for 6061 alloy, nor for other aluminum alloys. To fulfill this need, an experiment has been initiated in HFIR to measure radiation creep in miniature pressurized tubes of 6061-T651 alloy. It is hoped that the first data from this experiment will be obtained in the early months of FY 1995. Plans are under way to tackle the remaining outstanding issue for irradiated aluminum, namely the effects of a highly thermalized neutron spectrum (an important issue because much of the radiation-induced change in mechanical behavior of aluminum is caused by a precipitate of silicon) produced by thermal neutron transmutation of aluminum.

Aluminum is not the only material under study. The use of austenitic stainless steel is being studied for the primary coolant inlet adapter where erosion rates may be high. A potential problem of diminished fracture toughness in stainless steel because of excessively high rates of generation of helium in the steel by the strongly thermalized neutron spectrum around the adapter has been identified. Preliminary calculations indicate that localized radiation shielding can overcome this potential problem.

2.7.4 ANS Materials Information System

The ANS materials information system (AMIS) is providing hard copy and electronic versions of materials information for the ANS program. Initial efforts are providing data for those activities demanding immediate estimates of material properties. Future work will provide reviewed and approved versions to support final design and safety reviews.

An initial draft of an ANS materials databook (AMBK) was prepared. The AMBK includes (1) as-received information with no analysis or review and (2) draft analysis and summary versions of the as-received information. To date, a total of 270 pages of as-received technical data has been prepared on aluminum alloy 6061-T6, 6061-T651, alloy X-750, 304 stainless steel, and heavy and light water.

The commercial CenBase[®] (Information Indexing, Inc., Garden Grove, Illinois) materials data base has been installed on the Martin Marietta Energy Systems' Library compact disc, read-only memory (CD-ROM) network to augment the AMBK.

2.7.5 Fuel Plate Stability Tests and Analyses

Closely spaced arrays of fuel plates cooled by water flowing through the channels between the plates have long been used in research reactors. In some reactors where fuel plates are

cooled in this manner, plate structural failures have occurred (Stromquist and Sisman 1948; Beck, Brown, and Rainwater 1958; Beck 1958). To avoid such fuel plate failures in ANSR, it is important that reasonable analyses for predicting the plate deformation and stress under operating conditions are available. Such is the reason for the tests and analyses in this program.

This year a report, *Followup Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source* (Swinson et al., 1993), was published. The results of a series of four different flow tests on full-scale epoxy model involute plates are given in this report. The findings indicate that no unstable plate collapse from coolant flow is anticipated in the operation of ANS. However, the findings do indicate a need for a careful assessment of the stress and deformation induced in the operating plates.

Based in part on this experimental effort, design techniques for predicting the principal structural deformations and stresses in the fuel plates during operation are being developed. These analyses include creep and thermal-stress analyses, as well as the structural response of the plates to the coolant flow. In addition to the involute plates that are being analyzed with these techniques, alternative circular-arc plates are being examined for any potential advantage.

In conjunction with these analyses, a test section has been designed and fabricated to test the response of arrays of five active aluminum plates to flow velocities up to 30 m/s. Tests on these plates will be run in FY 1995. One objective of these tests is to assess the validity of the plate structural design techniques being developed. Another objective is to furnish a data base on circular-arc plates to augment the data base on involute plates. The final objective of the tests is to help assess any potential for flutter in aluminum plates that was not indicated in the earlier epoxy plate tests.

2.7.6 Fuel Plate Thermal Deflection

Finite element analyses have been performed on the proposed ANS involute fuel plates to

determine the hypothetical worst-case change in gap between the statistically hottest and statistically coldest fuel plates that result from temperature variations in the plates based on a revised fuel grading scheme. In comparing these results to the results from a previous fuel grading scheme, two observations were made. The first is that the maximum change in gap between the plates is increased. The second observation is that the location of maximum change in the gap is different.

2.7.7 Inner Control Element Structural Response

The inner control elements are relatively flexible because of their long unsupported length and small diameter. Therefore, turbulent axial and cross flow around them could result in vibration amplitudes that might cause concern about the safe operation of the reactor. The cross flow has been minimized by the inherent design, but several portions of the control elements will be exposed to turbulent axial flow. Large vibrations of the control elements could cause contact with the core inner cylinder, which would result in wear. Thus, control element stability tests and analytical model analysis must be performed to ensure that unacceptable vibrations will not occur.

2.7.7.1 Flow-Induced Vibrations

The three inner control elements are cylindrical rods confined to a small space inside the core. While initial analysis indicates that the operating conditions should be well below the threshold velocity for large divergent oscillations, a small steady-state oscillation will occur. Analysis showed that it is important that the lower roller supports for the inner control elements should be at least as stiff as the upper roller supports to reduce flow-induced vibration when the inner control elements are only supported by the lower roller supports.

2.7.7.2 Scram Performance Studies

All three of the inner control elements are extracted from the core until criticality is reached (moved upward about 600–700 mm) and then used for shim control. The rods are rapidly accelerated downward by springs and gravity forces when a reactor scram is required, and the rods are then stopped by a deceleration spring and a piston that acts as a damper.

The dynamic simulation of the scram process was improved by the addition of inertial and fluid force terms to the equations. Greater scram mass, consistent with the use of Inconel and Zircaloy for the support tubes, was studied. The studies showed that spring and damper forces were adequate to meet the preliminary scram performance criteria. The damper force vs distance profile was optimized to minimize the peak deceleration. Analysis of false scrams from very small extraction distances indicated the need for a bilinear deceleration spring.

Nonlinear analyses have been completed to determine if the bending of the inner control element caused by flow-induced vibrations would result in excessive lateral deflection during scram. The lateral deflections predicted by flow-induced vibration calculations were used as initial conditions to the application of axial dynamic loading caused by the scram deceleration. The analyses showed that the simultaneous occurrence of flow-induced vibration and scram would increase the lateral tip displacement by no more than 9%. It is concluded that scram will not cause excessive lateral displacement or buckling in the inner control element.

2.7.7.3 PACE Team—Control Element Test Facility

The control element test facility (CETF) will be used to evaluate flow-induced vibration associated with the inner control elements. The facility will also be used to determine operational characteristics and to determine operational reliability.

An interorganizational team, called the PACE* Team, was assembled late in May 1994 and colocated to design and construct the CETF. Design and analysis work during this report period has focused on the test stand components. Two test stands will be constructed: a flow test stand and an assembly test stand. In order to accommodate potential changes in the final control element design, the test stands are being designed with modular elements that are easily adjustable. Drawings for both the test stands have been completed, which provide for adjustable platforms that can be moved up or down in increments of 127 mm. These adjustable platforms, together with the use of shim plates, provide a test facility that can support the longest control rod design length envisaged (1470 mm of hafnium) with large values of margin.

2.8 COLD SOURCE DEVELOPMENT

Major effort over the past year has been centered on specific design issues identified by the baseline design that was reported last year.

2.8.1 Thimble Assembly

Each cold source moderator assembly produces a heat load of 32 kW, which comes from two roughly equal sources, neutronic and gamma heating in the liquid deuterium itself and heat generated in the walls of the vessel. For the proposed deuterium flow rate of 1 kg/s, such a heat load represents a temperature rise within the liquid of 5 K. The liquid is pressurized to raise its boiling point to a minimum of 27 K. Since deuterium freezes at 18.7 K, there is an operating envelope of about 8.5 K. At entry to the

* PACE is an acronym for the various organizations represented on the team including Procurement, Advanced Neutron Source Project, Crafts, Instrumentation and Controls Division, Engineering Technology Division, and Engineering Division.

moderator vessel, the temperature will be controlled at 20 K. Providing the necessary heat transfer coefficients between the vessel walls and the liquid deuterium will require careful flow control.

Figure 2.15 shows the configuration used for initial feasibility calculations. A thin walled inner sphere creates an annulus 2-mm wide through which incoming liquid is passed over the wall surfaces to remove the heat generated in the vessel. This flow raises the liquid temperature to about 22.5 K, after which it passes through the inner sphere, where its temperature rises to 25 K. Design of the internal baffling needed to produce the necessary velocity is very important as its mass must be as low as possible to limit heat production. The internal baffling must also meet other requirements of the system, such as a low flow resistance, which is particularly important during the cool-down and liquefaction stages. A computer model based on this configuration is included later in this report.

The moderator vessel is contained within a vacuum shell called the inner thimble. The

Fig. 2.15. Flow principles of liquid flow in cold source moderator vessel.

original concept is shown in Fig. 2.16. The conical shape of this thimble helps reduce the volume of heavy water reflector lost to the cold source, but it creates difficulties in positioning the liquid service point at the preferred highest point of the vessel. An alternative design is being evaluated in which the inner thimble is made up from two cylinders (Fig. 2.17). The outer thimble, which is effectively an integral part of the reflector vessel, would also require modification to accommodate this revised inner thimble. The design assumes that heavy water coolant would flow through the space between the thimbles, which addresses a separate cooling problem. A summary of the neutronic analyses of various configurations is included later in this report.

2.8.2 Safety and Hazardous Gas Handling

The safety of the cold source system design depends on four main features.

1. *Separation of deuterium from atmospheric air under all operational conditions.* This issue is addressed by double containment of the deuterium system, including vacuum-insulated items and the vacuum equipment itself.
2. *Pressure relief for parts of the system.* The main loop, the vacuum envelopes, and the outer inert gas containment are all fitted with rupture discs. These and any other ventilation points are piped to a large double-walled containment vessel. This provides sufficient volume to limit the maximum pressure to 0.12 MPa. Figure 2.18 is a diagrammatic representation of the system.
3. *Containment of possible leaks.* To limit the effects of a leak occurring during operational maintenance, it has been recommended that all components with a leak potential should be contained in a safe room. If this design change is approved, it will be designated a restricted area, and all electrical equipment

Fig. 2.16. The original thimble assembly cold source concept.

Fig. 2.17. Alternative thimble assembly cold source concept.

Fig. 2.18. Diagrammatic representation of the cold source pressure relief system.

within the room will be made intrinsically safe. Sensors will warn of any leakage and will switch in a closed circuit cleanup system that incorporates a catalytic unit to clear the room of deuterium. The deuterium gas will be reduced to heavy water by the catalyst and transferred to molecular sieves. The sieve boxes are portable and will subsequently be regenerated and the tritium removed in the detritiation plant.

4. *Inventory handling.* Gas will be transported to and from the cold source systems by portable metal hydride beds. The design allows a full inventory of gas to be transferred to the large ventilation vessel at a pressure of 0.2 MPa. This operation would leave the deuterium loop under vacuum and ready to be filled with inert gas. The hydride beds would then remove the gas down to vacuum level ready for a replacement charge. This gas can then be returned to the loop, restoring the vent tank to vacuum.

2.8.3 Cold Source Neutronics

Heat loads were modeled for a number of moderator and thimble configurations. Different material combinations were examined to provide data from which the best optimized design could be configured. The geometry shown in Fig. 2.17 represents the most neutronically efficient option so far, largely on account of the reduced loss of heavy water around the moderator vessel.

2.8.4 Computer Modeling

A major program of computer modeling has been started by the computer applications group. A moderator vessel model was generated using the FLOW-3D code (Fluent 1993) and used to examine flow characteristics and calculate heat transfer coefficients. The configuration described earlier was used for this exercise, which calculated velocities of about 15 m/s

at the top and bottom of the vessel and almost 5 m/s at the centerline where the cross-sectional area is maximum. Subcooled boiling was indicated, but subsequent bubble evaluation calculations indicated a diameter of only microns, which would recondense in the return leg of the cold loop. This work was preliminary, but it demonstrated the feasibility of removing these large heat loads using basically simple design principles.

2.8.5 Finite Element Stress Analysis

The Numerically Integrated Elements for System Analysis (NISA; Engineering Mechanics Research Corporation 1992) was used to compute stresses produced in various cold source and moderator vessel configurations. It became clear that the moderator vessel service pipe must enter normal to the surface and not at an angle as was originally planned. The penalty of the original design is a need to increase significantly local wall thickness at the pipe-vessel junction. Design iteration studies using NISA models resulted in a normal pipe intersection design that requires only a very limited thickness increase. It was demonstrated that all noncircular thimble geometries suffered from high stress concentrations.

2.8.6 Cryogenic Test Facility

Construction of a liquid nitrogen cryogenic test loop was initiated and nearly completed this year. The initial use of this facility will be the testing of the liquid deuterium circulators. The working loop will be pressurized to 0.4 MPa to create an operating envelope between 80 and 85 K, which is within the saturation curve for nitrogen. The circulator is prototypical of those proposed for the final system but fitted with a special impeller to allow the circulator to run at normal speed and load with liquid nitrogen. Tests will be made to determine the operating characteristics of the circulator and to demonstrate its long term reliability. The system

includes data acquisition for extended test runs and has been designed as flexibly as possible to offer a variety of future applications.

2.8.7 Cold Source Test Program Plan

During this report period, a cold source test plan was developed. The plan consists of individual component and system tests designed to provide a validation of the full cold source system. Individual test programs are planned for all key components of the cold source and will be used to provide early proof-of-concept data important to the development of the cold source design. An overall test of the full cold source system would be desirable but impractical. Instead, three individual (but interlocking) tests will be carried out, which, together, will represent the operation of a fully loaded cold source. The elements of this proposed system validation program are listed below.

1. A true scale model of the moderator vessel will be heated, possibly by induction coils and liquid deuterium circulated through it at the design flow rate. Heat will also be applied to the liquid deuterium at an intermediate point in its passage through the vessel to simulate neutronic heating of the liquid itself. The total heat applied will be prototypic of the real system. During the test, temperature mapping of the outer surface of the vessel will indicate any possible areas of inadequate cooling. This test program will be carried out in a large vacuum facility and will be able to simulate refrigerator or circulator failure and demonstrate the competence of the safety systems.
2. A full-sized test loop will also be set up using a refrigerator of about 1–2 kW capacity, which is sufficient for all normal static heat loads. The objective is to demonstrate an ability to fill and subcool the system with deuterium and to subcool it to operating temperature.

3. As part of the ANS facility start-up testing program, an integrated test of the installed refrigeration system will be performed with a dummy heat load to provide final validation that the system meets design requirements.

2.9 BEAM TUBE, GUIDE, AND INSTRUMENT DEVELOPMENT

Assessment of flux depressions resulting from perturbation effects caused by the beam tube structures has continued. Some relevant results are reported in the experiment system design section of this report (Sect. 3.3). The NIST group has used an independent technique to continue evaluation of the neutron optical filters discussed in last year's progress report. The results agree with the earlier Monte Carlo analysis and confirm that the idea will be particularly valuable as a necessary means of improving signal-to-noise ratio if use of a lower flux must be accepted, for example, through the use of lower enriched fuel.

The investigation of ultracold neutron production methods has been completed at the University of Rhode Island. That study (Crow and Steyrl 1994) indicates that an axial reflecting neutron turbine is the method of choice. Studies continue on the use of Liouville's theorem to optimize the very cold neutron feeder guide system for the turbine.

New results obtained from France on the stability under neutron irradiation of supermirror guide coatings provide good news for ANS; the supermirror guides should last the lifetime of the facility even with ANS magnitude fluxes.

Lessons learned from the Brookhaven TRISTAN accident investigation (DOE 1994) have been incorporated into the isotope separation-on-line (ISOL) conceptual design.

2.10 HOT SOURCE DEVELOPMENT

The design of the hot source requires knowledge of the thermal conductivity of the

moderator insulation materials. High temperature thermal conductivity data of candidate insulation materials are not available, so efforts are in progress to obtain such data, which will enable development of thermal conductivity models for the candidate materials.

Two carbon bonded carbon fiber (CBCF) materials and one graphite felt material were chosen for initial evaluation. One of the CBCF materials was manufactured by the ORNL Metals and Ceramics (M&C) Division, while the second was supplied by UCAR Carbon Company. The WDF-graphite felt was supplied by National Electrical Carbon Corporation. The conductivity of these materials depends on the gas environment and heat treatment (conditioning) temperature, in addition to the operating temperature. The test matrix has therefore included tests in helium (design basis) and argon atmospheres and in vacuum with material heat treatment at temperatures of 1873, 2300, and 3000 K.

Two subcontracts have been placed. One is with Southern Research Institute (SRI) where thermal conductivity tests are being performed at temperatures up to 3000 K in a helium atmosphere using a radial inflow technique. A second subcontract was placed with the Thermophysical Properties Research Laboratory (TPRL) where thermal diffusivity measurements are being made on these materials using the laser flash technique to obtain thermal conductivity. The measurements at TPRL will be made at temperatures up to 2500 K (up to 2900 K on a best effort basis) in vacuum. In addition, testing is also being performed at ORNL at temperatures up to 2300 K in helium using the laser flash technique.

Initial results from testing at SRI show some anomalous material behavior at temperatures above 2600–2800 K. However, preliminary results at temperatures below this value, taken both at ORNL and SRI, have indicated that the CBCF material has better insulation properties than does the UCAR material. Efforts to resolve the observed anomalous behavior are continuing.

2.11 NEUTRON TRANSPORT AND SHIELDING

2.11.1 Subpile Room/Coolant Inlet Module Calculations

A two-dimensional discrete ordinates transport code (DORT; Rhoades and Childs 1988) of the control rod guide assemblies between the bottom of the reflector vessel and the subpile room was developed in order to provide estimates of the expected dose rates in the subpile room. The source for this calculation was an angle-, energy-, and space-dependent boundary flux from a DORT R-Z calculation of the reflector vessel internals, including the effects of photoneutron production in the D₂O. The initial DORT calculation of the geometry extending to the subpile room did not include the effects of the photoneutron production. Since the control rod guide assembly below the coolant inlet module does include D₂O, the calculation was repeated in order to include photoneutron effects. The calculated total neutron flux in the top area of the subpile room was increased by approximately three orders of magnitude when the photoneutron production was included. The gamma flux, as expected, showed little difference between the two calculations.

Because of interest in the radiation damage to the coolant inlet module, the DORT model, which began at the bottom of the reflector vessel adjacent to the inlet module, was modified by extending the model upward approximately 900 mm. A new boundary source was extracted from the DORT calculation of the reflector vessel internals region, and the DORT calculation of the extended model was run with the new boundary source. The calculated neutron fluxes in the inlet module were higher than would be acceptable in practice, and, since it is not feasible to include shielding inside the CPBT, a shield consisting of an ideal black absorber was modeled just outside the CPBT, near the bottom of the reflector vessel. DORT calculations with and without this shield show that the ideal shield would reduce the thermal flux in the inlet module

flange at the bottom of the reflector vessel by about two orders of magnitude. The flux is dominated by the thermal part of the spectrum, and the reduction provided by the ideal shield is within the range required to ensure an adequate lifetime for the inlet module. Figure 2.19 shows the DORT geometry model and neutron isoflux contour plots in that geometry with and without the ideal shield in place. The next step in the analysis will be to perform calculations using actual shield materials and a more detailed model of other design features, such as the core support structure.

2.11.2 Analysis of the Beam Tube Shield Plugs

Surrounding a typical horizontal beam tube as it passes through the biological shield is a fixed, semipermanent, beam plug assembly and a movable beam tube shield plug or shutter assembly, shown in Fig. 2.20, which rotates about a horizontal axis parallel to the beam but 300 mm below it. Axially, this 1220-mm-thick shutter assembly consists of eleven alternating layers of steel and Benelex 70[®] (International Paper Corporation, Chicago, Illinois), an organic-based neutron shield. The shutter assembly may be rotated into any one of five positions, a closed position where the shielding material directly blocks the neutron beam, and four operating positions in which any one of several different collimators would be directly aligned with the beam tube, allowing the neutrons to pass through a corresponding hole in the spacer shield downstream of the shield plug.

With the shutter in the open position, the neutron dose rate in the beam downstream of the spacer shield, just outside the biological shield, would normally be 2×10^6 rem/h, while the gamma dose rate there would be 1.8×10^5 rem/h. Therefore, much of the work this year focused on the determination of the dose rates at this location with the shutter in the closed position. While a solid shield plug shutter assembly would have provided more than adequate shielding, the presence of the large collimator holes through the shield plug (just

Fig. 2.19. Thermal neutron flux contours in transport code subpile room geometry ($\text{m}^{-2} \cdot \text{s}^{-1}$).

366-mm away from the axis of the beam) were initially a major source of concern.

With the shutter in the closed position, most neutrons will scatter off the hot end of the shield plug and diffuse radially to the nearby holes, where they will preferentially stream axially through those penetrations. Subsequently, the neutrons will scatter off the hot end of the spacer shield and diffuse radially toward the large hole in the spacer shield, through which they may then escape.

A series of transport calculations was performed to determine the scattered boundary flux at the hot end of the shield plug, while a second set of calculations was performed to determine the transport of neutrons through the adjacent holes. From this, the boundary flux at the shield plug/spacer shield interface was determined, and a third set of calculations was performed to analyze radiation transport through the hole in the spacer shield. From these analyses, it was determined that the neutron

Fig. 2.20. Conceptual design for a standard beam-tube biological shield penetration.

dose rate downstream of the spacer shield with the shutter closed would be between 5 and 13 mrem/h (depending on the amount of space between the fixed beam plug assembly and the hot end of the movable shield plug assembly), and that the corresponding gamma dose rate there would be between 3 and 19 mrem/h.

2.12 INSTRUMENTATION AND CONTROL SYSTEM DEVELOPMENT

2.12.1 Self-Powered Neutron Detectors

Self-powered neutron detectors (SPND) have been evaluated and selected for use in one of the protection systems and for control measurements. SPNDs are widely used to map core flux in commercial power reactors. Most neutron-sensitive type detectors are

characterized by slow response and/or poor gamma discrimination. For mapping purposes, the slow response is not important, but for use in the reactor protection system (RPS), fast response is essential. A new type SPND has been developed by Atomic Energy of Canada, Limited (AECL) for in-core use in Canadian deuterium uranium (CANDU) reactors, that has both fast (essentially prompt) response and a favorable ratio of neutron-to-gamma sensitivity. This new detector type has an Inconel emitter and Inconel sheath and cable. The insulation used is an oxide of either magnesium or aluminum.

Construction is similar to high temperature thermocouples and results in a rugged, long-lived instrument. The detectors can be made with diameters ranging from 1.5–10 mm and with any length. A typical detector may have an outer diameter of 3 mm, with a lead section diameter of 1.5 mm. Currently, the plan is to use such detectors with an active length of about

1 m, approximately the same as the ANS core height. Typical sensitivity of a detector of these proportions would be $2.2 \times 10^{-25} \text{ A} \cdot \text{m}^{-1}$ per unit of neutron flux, where flux is in units of $\text{m}^{-2} \cdot \text{s}^{-1}$. If located parallel to the core and at a radial distance of 1.4 m, the output would be about 1 μA at full power. The neutron-to-gamma ratio at this position is at a maximum and greater by a factor of about 10^3 than at the chamber locations in the light water pool.

It has been proposed to locate clusters of two or three detectors in each of four thimbles spaced around the circumference at a radial distance of 1.4 m from the core centerline. The thimbles would be approximately 25-mm diam. The gamma sensitivity is only a few percent of the neutron sensitivity, and lead shielding will not be needed. At the planned locations, the thimbles may be operated dry or filled with light or heavy water with no appreciable effect on reactivity. Detector life at the proposed location should be 5 years or more. Sensitivity increases slowly with irradiation resulting from breeding of ^{59}Ni from ^{58}Ni , then decreases ultimately because of the depletion of ^{58}Ni . At the predicted flux in ANS, the sensitivity will be about the same as first used after 20 years. SPNDs located in the reflector will provide needed diversity from the conventional ionization chambers located in the light water pool.

2.12.2 Revised Control Rod Worth Requirements

A series of ANS Dynamic Model (ANSDM); calculations were performed to redefine the initial rod acceleration requirements using the most current values of rod worths and neutronic parameters (March-Leuba and Ibn-Khayat 1990). Recent calculations indicate that the neutron generation time is very sensitive to the amount of neutron absorption in the reflector region; thus, as described in Sect. 2.5, a conservative value of 0.5 ms has been used to define control rod requirements. Using this value and a scram rod differential worth of 50 mm/\$ (assumes one failed rod at the worst point in the cycle),

the required reactivity insertion rates have been calculated. To perform these calculations, the ANSDM simulation used a fixed reactivity profile in time, which included both the positive reactivity accident and the negative reactivity inserted in response to it by the protection system. This reactivity profile is produced in two steps. First, an ANSDM calculation is performed to determine the time at which the rod would start moving. (In this way, the protection system response and the reflector D_2O delays are included in the calculation.) Then, the response of the protection system is superimposed on the positive accident reactivity, and the final calculation is performed. All the calculations were performed at the 95% uncertainty level because a large (>0.5) reactivity perturbation is an extremely unlikely event at ANS.

The results of these simulations are the minimum required times to insert \$1.0 worth of reactivity as function of the postulated reactivity step amplitude (Table 2.3) or the reactivity ramp rate (Table 2.4). These analyses have confirmed that the present inner rod acceleration requirement of 70 ms to insert \$1.0 is valid. With a differential rod worth, at the worst point in the cycle of 50 mm/\$, this requirement translates to an approximate acceleration of only about $\sim 20 \text{ m/s}^2$ (twice gravity).

Table 2.3. Scram requirements as function of postulated reactivity step

Reactivity step (\$)	Time to insert \$1 (ms)	Required acceleration (m/s^2)
0.5	1000	0.1
0.6	290	1.2
0.7	135	5.6
0.8	75	19.1
0.9	40	63.7
1.0	25	163.3

Table 2.4. Scram requirements as function of reactivity ramp rate (limited to \$1.0)

Ramp rate (ms/S)	Time to insert 1S (ms)	Required acceleration (m/s ²)
0	25	163.3
25	26	145.7
50	40	63.8
75	67	22.5
100	127	6.21
150	319	0.98

2.12.3 Revised Control Rod Drive Requirements

Three significant changes are being made to the baseline requirements for inner control rod performance. These are: (1) elimination of the fast-insertion air motor, (2) elimination of hard limits for the regulating function, and (3) combination of the regulating and shim drive functions into a single motor with an appropriate drive speed. These changes will result in considerable simplification of the drive mechanism, resulting in higher reliability and easier maintenance. The baseline design was patterned after the very successful performance of HFIR, recognizing the many similarities between HFIR and ANS. However, the slower neutron generation time of the ANS allows some reoptimization of rod performance.

The functions of the air motor specified in the conceptual design report are to assist a fast restart following a scram, because the control rods must be fully inserted before the rods can be relatched, and to help deal with transients that require a fast power reduction (e.g., loss of flow because of trip of one or more of the main circulation pumps). The slower generation time and, thus, slower neutronic response time of ANS allows faster operating speeds of both the servo and shim actions. With these higher velocities, the air motor is no longer needed to perform a fast restart. Even if the rods were

fully withdrawn at end of cycle, it would take only 2.5 min to insert the drive fully and relatch it, which is a reasonable time to attempt a fast restart.

In addition to the air motors, the baseline design requires that the control-rod motion be limited to a value equivalent to a reactivity of $\pm 40\delta$ when driven by the automatic control system. This limit is enforced in hardware by allowing only ± 17 mm of motion to the servo mechanism. The need for these hard limits has been reevaluated based on simulations with the ANSDM code, which reveal that, even if the servo should run away at its maximum speed, the rods would scram before the ± 17 -mm limit is reached. The advantage of eliminating the hardware limits is that, without them, the servo and shim functions can be accomplished by a single motor. The original conceptual design required two separate motors with a relatively complex mechanism to allow for independent motions. Combining the functions mechanically with a single motor while electronically maintaining the appearance of separate drives for better operator interface has been proposed. This scheme, known as a "traveling limit switch," has been used successfully on the Oak Ridge Reactor (ORR) and the Bulk Shielding Reactor (BSR).

A series of ANSDM simulations has been performed to determine the minimum speed required by the combined servo-shim drive in the absence of air motors. These simulations indicate that the most demanding transient the control rod must handle is a hypothetical transition to pony flow from full power and full flow conditions without scrambling. The control rod must move at a minimum velocity of 10 mm/s to avoid scram on greater than 115% power-to-flow ratio. Thus, the minimum velocity requirement in the original baseline design of ~ 15 mm/s still appears appropriate. With this velocity, the CHF ratio never approaches its 95% uncertainty safety limit even for this limiting transient. The proposed minimum speed requirement of 15 mm/s is equal to the servo speed in the baseline design but is twice the baseline shim speed. Increasing the shim speed might conceivably have a deleterious effect on

the response to “start-up accidents” in which the inner rods are assumed to run away because of a control failure during the approach to criticality, but this transient has been analyzed with ANSDM; the results indicate that the proposed 15 mm/s shim speed is acceptable, and “start-up accidents” are of little or no consequence in ANS. The results are summarized in Fig. 2.21.

2.13 FACILITY CONCEPTS

2.13.1 Reactor Components Test Facility

Testing requirements for the reactor components were identified, and a preliminary design of the test facility was developed. The following facility tests have been identified:

(1) seal tests, which involve testing of the CPBT seal and experiment tube seals, and the locking ring; (2) thimble tests, including external pressure, pressure surge, thimble installation, and face seal tests; and (3) outer control rod (OCR) tests, involving a single OCR test and a full scale OCR test. Each of these tests is composed of multiple elements addressing component design issues, verifying analytical models, evaluating performance parameters, and providing prototype demonstrations.

The reactor component test needs were translated into a series of test facility requirements. Preliminary layouts for the facility, including system schematics, were drafted. Schedules, cost estimates, resource requirements, risk evaluations, and alternate site considerations were performed. Presentation to the ANS Test

Fig. 2.21. Reactor power during a start-up accident caused by uncontrolled rod withdrawal at 15 mm/s peaks at only ~10% power, even if the primary scram fails.

Facility Review Board is scheduled for the first quarter in FY 1995.

2.13.2 Inner Control and Shutdown Rod Tests

Integrated operation and response of the complete full scale inner control rod system will be tested in the control element test facility, which is part of work breakdown structure (WBS) 1.1.7. However, certain separate effects tests are performed under WBS 1.1.13 to provide early proof-of-principle testing of key control and shutdown rod concepts. Four tests of this type have been identified, and the work performed this year on each is discussed below.

2.13.2.1 Latch Release Magnet Test

The latch release magnet test was completed in the first quarter of FY 1994. The test characterized performance of the latch release magnet and determined its load carrying capacity and response time. Analytical tools used in the latch magnet design were verified. Because of the small space for this component in the reactor, a very compact winding design is necessary. Analysis had shown that the baseline latch release magnet was marginally able to support the applied loads.

Candidate materials were selected and full-scale electromagnets were fabricated. The magnets were mounted in a standard tensile testing machine. Various current levels were applied to the magnet windings to establish current/load relationships. With a fixed applied load, a square wave current was applied to the magnet windings, and the current period was increased until the magnet released. This established the response time between loss of current and magnet release.

A cylindrical magnet of 2.5% silicon steel, with an air gap between armature and yoke to ensure consistent release times, met the holding force and time release requirements. Figure 2.22 is a photograph of the magnet test stand.

Fig. 2.22. Advanced Neutron Source latch magnet test stand.

2.13.2.2 Latch Wear Test

A conceptual design was developed for a wet wear test of the latch. The test plan was presented to the test facility review board in the first quarter and was approved pending results of preliminary wear tests being conducted by the ORNL M&C Division. The use of traditional hard-surface materials, such as Stellite® (Cabot Corporation, Kokomo, Indiana), which contain high levels of cobalt, has been restricted because of activation levels.

Test latch component materials will be determined by the preliminary wear tests. A computer controlled tensile test device will apply loads and motion that simulate the normal latched, scram, and relatching scenarios. This test device will be able to take the hardware

through operation and scram cycles, and to perform life-cycle tests. The latch hardware will be immersed in water of the appropriate pH to develop wear vs time characteristics of the ANS latch mechanical components. The design of this test is expected to be completed in FY 1995.

2.13.2.3 Lower High Pressure Seal Test

The purpose of the lower high pressure test is to determine the seal leak rate characteristics and provide proof-of-principal for the concept of the buffered high pressure seal of the control rod system where it passes into the subpile room.

A letter was sent to potential seal manufacturers requesting an expression of interest in designing and testing a seal concept for the inner control rod seal design. Replies were received from three companies. Two expressed interest and one returned a no interest reply. These two companies will now be pursued to perform the design and carry out the tests.

2.13.2.4 Corrosion Products Buildup Test

Before preparation of a test plan, a literature search and personal interviews were performed to obtain information relating to any control rod latch problems on HFIR, ORR, or other DOE reactors caused by the buildup of corrosion products. No evidence was found of corrosion product problems in any other release latch mechanisms, and so corrosion product buildup testing on release latch components has been dropped from the testing plan.

2.13.3 Core Pressure Boundary Tube Seal Test

In addition to the above work on the inner control rods, a static CPBT seal test was developed and was approved by the test facility review board. The test hardware was designed and procured this year and consists of concentric cylinders joined by a prototypic lower CPBT seal. The device, as shown in Fig. 2.23, will be statically pressurized to characterize leak rates.

2.13.4 Flush/Purge Lock Test

A conceptual design for a test of the flush/purge lock used for transferring fuel from the reactor pool to the storage pool was prepared and presented to the test facility review board. The purpose of the test was to characterize the efficiency of flushing tritium-contaminated heavy water from the fuel element assembly. The board's action was to delay the test until additional, more accurate information is known about the lock geometry.

2.13.5 Heavy Water Detritiation Test

Conceptual design began on the combined electrolysis and catalytic exchange (CECE) demonstration. This test is to operate certain key components of the ANS heavy water upgrade and detritiation facility at as close to actual conditions as possible. Components to be included in the test are the electrolysis cell, the catalyst, the trickle bed recombiners, and the CECE column control scheme.

Fig. 2.23. Advanced Neutron Source core pressure boundary tube seal test stand.

DESIGN 3

3.1 SYSTEMS ENGINEERING

3.1.1 Predictability

Predictability and its relationship to availability and the ANS operating strategy were investigated because of the importance of predictability to the scientific community, the desire to include predictability and availability requirements in the next revision to the *Plant Design Requirements* (PDR; Booth 1994), and the need to proceed with activities required by the reliability, availability, and maintainability (RAM) plan for the project.

Simply stated, the predictability requirement from the scientific community is to provide lots of neutrons when, and only when, they are needed for research. From the scientists' perspective, this requirement is second only to the neutron-flux-intensity requirement and the "equal-to-HFIR" goal for irradiation capabilities. Predictability can be stated mathematically as the conditional probability that the facility is producing neutrons given that it is scheduled to produce neutrons. A value of 0.85 was specified by the scientific community for this conditional probability (Kohn 1993). Predictability depends not only on the performance of the facility hardware, a traditional RAM area, but also on the operating schedule and operating philosophy of the facility. The aim of the study was to understand the implications of the above predictability requirement for traditional RAM

requirements, such as availability, and to make sure that the facility design and operating strategy could achieve the PDR requirements.

The first task undertaken was to investigate the availability and predictability of HFIR (Merryman, Ballon, and Christie 1994) and ILL (ILL 1994). This study led to practical bounds for predictability of a research facility based on its operating philosophy and availability. It also led to estimates of availability and predictability performance for ANS based on the historical performance of HFIR and ILL. A significant factor in ANS performance compared to HFIR and ILL performance is that ANS has about a 20% availability disadvantage because of its shorter core life and higher neutron flux. The higher flux leads to more frequent, time-consuming replacements of components near the core as a result of the high neutron doses they receive.

A second emphasis of the study was simulations of ANS facility operation using a software tool based on Monte Carlo techniques. The code simulated four major phases of facility operation: neutron production, refueling, normal maintenance, and replacements of components near the core. Statistical parameters input to the code included scram and forced-shutdown probabilities and the mean times (and associated probability distributions) for refueling, normal maintenance, and replacement activities. Output of the code included availability, predictability, and the probability of starting and stopping neutron production on schedule. Through these

simulations, the bounds discussed above were verified, and estimates of predictability were made for a wide range of operating assumptions and parameter values. These parameters included the mean time for refueling and its associated probability distribution, the core lifetime, the scram and forced shutdown rate, and the degree to which including "cushion" or "hold periods" in the schedule reduces any mismatch between scheduled and actual operation, thereby increasing predictability. Many of these simulations also provided information that may aid in making decisions about core life associated with the enrichment study.

The study culminated with the formulation of design and operating requirements included in revision 5 of the PDR and in draft allocations of availability requirements to hardware systems of the facility. This work will be documented during early FY 1995.

3.1.2 Reliability, Availability, and Maintainability

Wayne Bonnett, Director of the Quality and Reliability Division of DOE-Oak Ridge Operations (ORO), led an independent assessment of the project's RAM activities and plans. The seven-person assessment team consisted of RAM experts, DOE-ORO managers, a representative from HFIR, and the RAM leader from the continuous electron beam accelerator facility (CEBAF) project. Formal presentations were given to the assessment team by ANS staff, and additional informal presentations were made at the team's request. The debriefing by the review team at the end of the week-long review was favorable.

Preparations by the ANS staff for the review led to an appreciation of recent project accomplishments that will contribute to reliable facility operation. A self-assessment of the overall RAM program demonstrated that RAM planning covers the entire project; implementation of the RAM plan is on schedule; and help from an industry team is needed as soon as possible to move the RAM program

forward in the most cost-effective way. Other noteworthy accomplishments included performing availability assessments of selected systems that could significantly affect neutron production. These were supported by Reliability and Performance Associates (Knoxville, Tennessee), a subcontractor to the project. At the request of the design team, additional reliability studies determined the appropriate redundancy and system configuration of several possible cooling system arrangements. Work sponsored by the Instrumentation and Controls Division of ORNL led to a dynamic maintenance management methodology for the project. Finally, to ensure meeting design requirements for availability and predictability, an initial allocation of inherent availabilities has been made to all the plant systems. This allocation will be refined during the next year.

3.1.3 Maintenance, Operation, Research, and Radiation Zoning

The ANS conceptual design offers a successful departure from any previously designed research reactor by providing a clear distinction and separation of the various activities that take place in the facility and by integrating many of the security, maintenance, and as low as reasonably achievable (ALARA) controls into the physical structure. The conceptual design was preceded by a careful analysis and prioritization of numerous criteria that laid the foundation for a user-friendly facility. The ANS buildings and site planning requirements data base (a flat-file data base) were developed in order to allow the designers to capture, store, and manipulate design and user requirements that supported and/or dictated the design. As the ANS systems are being defined, in preparation for Title I, methods of checking and refining the maintenance and operation requirements for those systems and their components began to be explored. A maintenance and operation integration (MOI) group was formed. This active group laid the foundation for the maintenance, operation, research, and radiation zoning (MORRZ)

activity with a goal to identify multiuse areas and resources of ANS during maintenance and operation so that proper planning and design activities can eliminate conflicts. "Resources" as used here refers to building zones, aisleways, lay down areas, equipment storage, shops, routing for waste management, handling equipment (such as cranes, hoists, and casks), consumables, and personnel. The approach is to develop the sequence of events that must take place and the conditions that must exist for the different plant states and during the transition between states in order to identify areas of potential conflict and to create a mechanism that captures all the information in an easily retrievable and useful format.

This newly captured information is weighed against many other issues that affect the developments of those systems and the ANS design as a whole. For the designer of the facility, the MORRZ process and associated manipulation tools and data repository provide an opportunity to learn which system maintenance and operation factors affect spatial design requirements. MORRZ will also allow the designer to identify systems' conflicts, duplications, and omissions, as well as to validate system design interfaces. MORRZ will also provide a better understanding of how each system fits within the "global" ANS design and will effectively monitor the changes that occur during the design process.

Various tools are used or being developed to aid the MORRZ process. These tools include scheduling software that allows preparation of process logic charts in conjunction with an interface assessment check list developed by the ANS team. A requirements document for the MORRZ application model, which captures the information generated by the activities mentioned in a multitude of data tables linked to each other by a relational data base model, has been drafted and reviewed.

Another part of the MOI activity is an aggressive effort to define a plant state diagram that also describes systems' states. This effort is crucial to the MORRZ activity because it places every maintenance and operation activity

on a time line that considers the dependency of systems on other systems and on the plant state and, thus, allows consideration of the aggregate, redundant, or conflicting requirements on resources at any given time. An activity analysis effort that analyzes and formally defines what functions are required to meet the needs of the MORRZ has begun. This process will lead to prototype development. As this prototype is developed, the data model, functional analysis, and formal requirements will continue to be refined and will lay the foundation for the sets of data bases. Graphical user interfaces (GUIs) will be developed for the various disciplines to facilitate data query and manipulation from anywhere in the system.

3.1.4 Definition of the Roles of the ANS Operators

The ANS team is committed to a design that meets the requirements of both the DOE and the U.S. Nuclear Regulatory Commission (NRC), such as Chapter 18, "Human Factors Engineering," of Nuclear Regulatory Guide (NUREG)-0800 (NRC 1984). In order to conduct effectively ANS task analyses, information based on the definition of the roles of reactor operators (ROs) is also necessary. This information was gathered by means of structured interviews.

Initial efforts were focused only on the RO position and involved only activities associated with normal operation. [Other positions (such as the senior reactor operator, auxiliary operator, shift supervisor, and shift technical advisor) and off-normal operations will be addressed in later studies.] A questionnaire was developed for each of the five technical areas—(1) reactor operations, (2) refueling operations, (3) maintenance, (4) cryogenic operations, and (5) detritiation operations. Special consideration was also given to the RO's interactions with research support personnel.

The groups interviewed included representatives from the ANS design team, HFIR operations management, and HFIR operators. A nuclear power facility that deals with detritiation

issues was also contacted for information on detritiation operations. The interviewees from the ANS design team were engineers involved in the design of specific reactor systems. In particular, design engineers in the following areas were contacted: instrumentation and controls, detritiation, cryogenics, cooling systems, core components, refueling, irradiation facilities, and beam facilities.

Preliminary results support the role of the ANS RO as a high-level system supervisor. The traditional RO role of analog system monitoring and system supervision will be considerably enhanced by the use of digital technology. Some examples of this enhancement include the use of system interactive procedures, digital measurement and logging, and implementation of lockouts and tagouts. In general, the ANS RO's base responsibilities will be similar to those of an RO at other DOE research reactors. This position may still require significant manual interaction with the system in order to facilitate testing and calibration; however, at this time, such activities seem to be necessary and desirable to ensure the RO's systems awareness. The ANS RO will ensure the appropriateness and thoroughness of automated safety functions. Ensuring safe and efficient operation (in terms of neutron production) and maintaining the availability of the reactor and the neutrons for research may be the primary goals of the RO. Success in these areas will comply with technical specifications and investment protection concerns. The ANS RO will also communicate with many other personnel when coordinating special processes, such as cryogenics and detritiation.

Figure 3.1 describes the roles of the ANS RO in relation to their activities and plant states from a systems perspective. The small circles around the center are the RO's identified roles. The "tool box" in the middle of the diagram includes the human functions that the RO uses to accomplish the activities in the next outermost circle. The outermost circle depicts the various plant states in which the RO may be in any of the roles, may use several tools in the tool box, and may accomplish multiple activities.

Other results of this study include the following three issues. First, the relationship between experimenters and the ANS ROs requires an experiment coordinator who would function as a liaison between experimenters and the operations and maintenance staff of the ANS. Second, there is a general consensus that the RO's role needs further to integrate maintenance activities. Third, because of the distinct and complex activities associated with cryogenics and detritiation, ROs will likely not be able to provide efficient, detailed assistance in these areas. It was therefore confirmed that special process operators both for cryogenics and for detritiation need to be established, a refinement of the originally proposed auxiliary operator concept.

3.2 REACTOR SYSTEMS

3.2.1 Design Model

The process of changing from two-dimensional- (2-D-) based design drawings to three-dimensional- (3-D-) solid computer models was started. Assemblies are constructed by first building individual component models in a computer-aided design (CAD) workstation and then putting the models together. Interferences and other problems are discovered at an early design stage and can be corrected. This procedure results in better design integration between different systems. For complicated geometries, the 3-D model is much better for visualizing the part. The 3-D models can also translate easily into the surface modeling programs used to perform motion studies for insertion and removal paths.

A starting point for this process has been the construction, as one model, of the baseline reflector vessel. The different components within the vessel were constructed as separate models and assembled into the vessel.

Fig. 3.1. The roles of the Advanced Neutron Source reactor operators.

Fig. 3.2. Reflector vessel assembly.

Figure 3.2 shows a view of the reflector vessel assembly with most of the internal components installed. An example of one of the components that make up this assembly is the upper end of a typical hydraulic tube as shown in Fig. 3.3. This irradiation facility requires a central flight tube and two heavy water return flow paths at different pressures. The design

Fig. 3.3. Typical hydraulic tube.

shown provides this capability and is connected to the reflector vessel with a clamp that will use a single point drive for ease of maintenance. Similar designs have been used for all the irradiation facilities located in the vessel. Another component example is the outer shutdown rod array, which is shown in Fig. 3.4.

3.2.2 Pressure Boundary Material Evaluation

A significant design task for the ANS primary pressure boundary has been investigating the proposed material change for the primary supply vessel adapter (PSVA; Fig. 3.5). This is a flanged vessel that connects the primary coolant manifold to the bottom of the ANS reflector vessel (RV) and to the bottom of the CPBT. The proposed material change is from the present 6061-T6 aluminum to 316NG stainless steel. The advantage of 316NG stainless steel is a much higher fracture toughness than aluminum and, because of this, an argument can be made that it can be designed under the assumption that a double-ended guillotine break will not occur on the basis of fracture mechanics and testing.

In order to evaluate this proposed change, it has to be shown that the stainless steel's mechanical properties (primarily fracture toughness and strength) will not degrade below an acceptable level as a result of elevated temperatures and radiation and that the aluminum flange of the RV in contact with the stainless steel PSVA will not overheat above design limits.

Work on this task for the past year has primarily been in calculating flux levels at the PSVA location to evaluate shield concepts that most effectively reduce flux levels and implementing mechanical property changes for these flux levels through a literature search of available data. In addition, new thermal calculations have been made for the RV and PSVA flanges using estimates of the reduced heating rates that would exist with shielding in place.

At this time, the most effective of the shield designs evaluated is a flat disk inside the bottom of the RV (over the top of the PSVA) combined

Fig. 3.4. Outer shutdown rod array.

Fig. 3.5. Primary supply vessel adapter.

with a cylindrical part around the bottom of the CPBT, about 1 m in length (see Fig. 3.5). This combination shows a thermal flux reduction (at an energy level: $E \leq 0.625$ eV) of two and a half orders of magnitude (factor of 500) over no shield.

At this level of reduction, the literature survey has demonstrated that the reduced strength and fracture toughness of 316NG stainless steel over the 40-year life of the reactor should not be significant. Factors introduced because of radiation that degrades mechanical properties of the 316 stainless steel are the development of helium and transmutant carbon in the material and atomic displacements. The literature search has shown that at the reduced flux obtained by the above described shield design, the helium and carbon levels, as well as the atomic

displacements, are too low to affect significantly the mechanical properties of a 316 stainless steel PSVA at temperatures less than 150°C.

Initial flux calculations used ideal absorber properties. Additional analyses using material properties of actual shield materials and the flux depressing effects of the core support components are in process.

Additional thermal calculations with conservatively reduced heating rates have been performed for the PSVA/RV flange interface. There will be a reduction in heating rates in both the aluminum and stainless steel materials of the flanges because of the flux reduction effects of the shield. These calculations have shown peak temperatures to be approximately 70°C in both the aluminum RV flange and stainless steel PSVA flange.

Some calculations remain to define these parameters more precisely and to assess the impact of the neutron shield on the flux at the beam tubes and irradiation facilities. Results from both the flux and thermal analyses performed to date are encouraging for the change to a stainless steel PSVA in the conceptual design of the ANS.

3.2.3 Modular Refueling Concepts

The aggressive availability goals of ANS require that the refueling operations be kept to a minimum. Additional studies were performed on the refueling process. The time required for refueling is dictated by cool-down time of the core plus the time required to remove the components necessary to allow access to the fuel elements, install the necessary absorbers to keep the fuel elements subcritical, remove the fuel elements, install new fuel elements, and then reinstall the components to complete the assembly. The baseline design is such that there are six different components, shown in Fig. 3.6, that must be removed and reinstalled to accomplish the refueling operation. The removal of each one of them is a multioperation process. Combining several of these components into single modules could potentially reduce

the overall time required for refueling. In order to identify which components could logically form modules, a matrix was generated in which all of the components of the reactor assembly and the experiment systems were listed both on the vertical and the horizontal axes in the order in which they would be removed. For a specific component listed on the vertical axis, the components that must be removed to remove the selected component could be identified by an "X" marked in the column under the component requiring removal. Groupings of X-marked components on the horizontal axis for adjacent components in the vertical axis indicate potential modules.

Such analyses indicates that it may be possible to design a single module that could be replaced at refueling. It would include the fuel elements, the CPBT, the fuel support, the load stack, the transuranium rods, the closure elbow, the irradiation capsule assembly, and the shutdown rod assembly as shown in Fig. 3.7. This approach appeared to have many benefits. The most obvious benefit is the potential for shortening the overall refueling time by reducing the number of operations that must be performed, but this approach also would allow the changeout of key components every fuel cycle, thus allowing frequent inspection of those components. If the component showed no deleterious effects, it could be reinstalled on a subsequent cycle. There is also the possibility that, by having fewer components to remove, the remote tool head of the refueling machine could be simplified. Also, the fewer operations that are performed, the fewer opportunities there are for creating loose parts that could find their way into the primary coolant path.

The module approach does create some design impacts that will require evaluation. The size of the module would affect the envelope and lifting capacity of the handling equipment and, possibly, the size of the refueling transfer cell. This configuration also dictates that the load transfer paths be designed to be internal to the module. Such a design might make the CPBT part of the load path, which would complicate the design and analysis of the CPBT. There would

also need to be provisions for disassembly of the module while maintaining criticality control of the fuel. At the present time, the decision is to continue the design based on removing individual components, but the modular approach will be studied further in the meantime.

3.2.4 Evaluation of the Three-Element Core

In the most recent enrichment studies, three-element core configurations were considered as a means of increasing the core volume to allow the use of lower enriched fuels. In the course of the analysis, some other potential advantages of the three-element configuration were also studied by the ANS Project. If the power is held constant, an increase in the core volume may allow the use of lower enriched fuels while simultaneously increasing cycle length and operating at a lower power density. Six different arrangements of the three elements onto a single core assembly were identified. The nomenclature adopted for the different core configurations is a two-letter identifier in which the first letter indicates the size of the element located in the center of the configuration (Small, Medium, or Large) and the second letter indicates the location of the largest remaining element (Top or Bottom). Each of the arrangements was evaluated based on performance, impact on other components, and mechanical design advantages. Table 3.1 shows the different arrangements and the characteristics for which they were evaluated.

After careful consideration, it was determined that the "S,T" configuration had mechanical design and material irradiation capability advantages over the others. It provided a large volume for material irradiation and isotope production facilities. It also allowed all the in-core irradiation facilities to be located downstream of the fuel elements, thus minimizing potential flow perturbations and possible loose parts at the inlet to the fuel. Further detailed refinements to create elements with similar plate stiffness led to the dimensions shown in Fig. 3.8.

Fig. 3.6. Components removed during refueling.

Fig. 3.7. Modules removed with modular refueling option.

Table 3.1. Characteristics of different three-element core configurations

Arrangement	M,T	S,B	L,B	M,B	S,T	L,T	2 Element Core
Number of Load Paths	1	2	2	1		2	
Irradiation Volume Down stream (L)	11.7+14.6 = 26.3	11.7 + 9.5 = 21.2	29.2	14.6 +19.9 = 34.5	29.2	11.7 + 15 = 26.7	
Longest Continuous Irradiation Length (mm)	418	418	856	418	856	418	
Full Delta P Pressure Boundaries In Tension		1	1	0	0	0	
Full Delta P Pressure Boundaries In Compression	0	0	0	0	1	1	
Component Reactivity worth (%)	BOC -6.9 EOC -7.1	BOC -5.0 EOC -5.8	BOC -6.9 EOC -7.6	BOC -5.5 EOC -6.3	BOC -5.4 EOC -6.7	BOC -8.3 EOC -7.7	BOC -10.5 EOC -10.6
Peak Thermal Flux ($10^{19} \text{ m}^{-2} \text{ s}^{-1}$)	BOC 5.41 MOC 5.84 EOC 6.14	BOC 6.43 MOC 6.30 EOC 6.11	BOC 5.85 MOC 6.29 EOC 6.06	BOC 6.29 MOC 6.51 EOC 6.15	BOC 5.46 MOC 5.62 EOC 5.99	BOC 5.86 MOC 6.13 EOC 6.12	BOC 6.83 MOC 7.07 EOC 7.25
Thermal Flux at Beam Tube Tip ($10^{19} \text{ m}^{-2} \text{ s}^{-1}$)	BOC 5.08 MOC 5.46 EOC 5.51	BOC 4.33 MOC 5.13 EOC 5.32	BOC 5.00 MOC 5.44 EOC 5.67	BOC 4.59 MOC 5.19 EOC 5.75	BOC 4.58 MOC 5.14 EOC 5.24	BOC 5.48 MOC 5.74 EOC 5.50	BOC 6.11 MOC 6.45 EOC 6.46
Thermal Flux at Cold Source Center ($10^{19} \text{ m}^{-2} \text{ s}^{-1}$)	BOC 2.99 MOC 3.15 EOC 3.26	BOC 2.70 MOC 2.97 EOC 3.11	BOC 3.03 MOC 3.24 EOC 3.40	BOC 2.82 MOC 3.05 EOC 3.29	BOC 2.76 MOC 2.97 EOC 3.12	BOC 3.22 MOC 3.37 EOC 3.36	BOC 3.34 MOC 3.48 EOC 3.55
Advantages	Single load path that is outside the irradiation assemblies. No internal pressure boundaries.	Inner CPBT is short. Possible location for isotope targets which would not require removal each refueling.	Long volume for material irradiation capsules and isotope production targets down stream of fuel.	Inner CPBT is short. No internal pressure boundaries. Single load path.	No internal pressure boundaries in tension. Long volume for material irradiation capsules and isotope targets.	No internal pressure boundaries in tension	
Disadvantages	No provisions for thermal expansion of fuel plates. Inner CPBT is excessively long	Dual load paths with irradiation capsule assy between.	Dual load paths with irradiation capsule assy between.	No provisions for allowing thermal expansion of fuel plates.	Inner CPBT is excessively long. Dual load paths with irradiation capsule assy between.	Dual load paths with irradiation capsule assy between

Fig. 3.8. Modified S,T configuration of the three-element core.

As the volume of the core increases, the thermal flux in the reflector vessel decreases and so the thermal flux for the three-element core is lower than that for the two-element baseline configuration. Two additional modifications to the S,T configuration were investigated to minimize the performance (flux) impact of switching to the larger core volume. These were based on the S,T configuration. The first was the S,T-OL1 in which the upper element was moved down toward the centerline of the core to

create a 50% overlap of the upper and the center elements. The second was the S,T-OL2 in which the two upper elements are overlapped 100% as shown in Fig. 3.9. The 100% overlap case has the best neutronic performance (Table 3.2) and is the one selected for further neutronic analysis and design studies.

Fig. 3.9. S,T-OL three-element core configuration.

Table 3.2. Unperturbed neutronic performance for three-element, S,T cores

	Reflector peak thermal flux ($10^{19} \text{ m}^{-2} \cdot \text{s}^{-1}$)	HB ^a thimble tip maximum flux ($10^{19} \text{ m}^{-2} \cdot \text{s}^{-1}$)	Flux at cold source position ($10^{19} \text{ m}^{-2} \cdot \text{s}^{-1}$)
Nonoverlap	6.0	5.2	3.1
50% overlap	6.5	5.9	3.3
100% overlap	6.8	6.3	3.5

^aHB = horizontal beam (tube).

3.3 EXPERIMENT SYSTEMS

3.3.1 Beam Tube Optimization

It is well known that, for a vertically focusing monochromator that is fully illuminated by a source, the intensity (neutrons per unit height) at the sample position is independent of the source height but linearly proportional to the height of the monochromator (Currat 1990). The initial choice of elliptical beam tubes (200-mm high by 100-mm wide) in the CDR was made on the basis of maximizing ultimate counting rates for various experiments at specified resolutions, without accounting for the effect of such large beam tubes on reactor performance. To ensure the structural integrity of these large, noncircular tubes, thick walls (14 mm) were required, and a 3-mm-thick cooling jacket around the tubes was needed to provide a flux of heavy water to cool the thick walls. Neutronic models that allow an assessment of the effect of the beam tubes on beam tube current have now become available, and the cumulative effect of these features on the reactor performance and, therefore, on the neutron current density down the beam tube has been shown to be rather large. Thus, a reevaluation of the beam tube geometry is in progress.

Current work is evaluating the effect that decreasing the beam tube thimble height would

have on the neutrons incident at the sample position, for all of the instruments planned for the reactor hall. The neutron delivery system consists of a thimble with height less than 200 mm, followed by a tube section, a guide section with supermirror coatings on the top and bottom, and a vertically focusing monochromator, respectively. The final height of the delivery system before the monochromator is 200 mm. Parameters to be optimized for a specified thimble height and fixed distances from the source to the monochromator and from the monochromator to the sample are the lengths of the various pre-monochromator sections and the vertical height of the channel as a function of distance from the source. The acceptance diagram method as described by Copley (1990, 1991) is used.

When the beam tube has a significant perturbation on the source flux, and for small samples (less than the image height), the number of neutrons on the sample can actually be increased by decreasing the source height (smaller thimble, less perturbation, and increased source flux). The situation is more complicated when a short guide is inserted before the monochromator, but this general conclusion is still valid. This is the effect that is being studied in seeking an optimized design.

The outward-bound current density ($\text{m}^{-2} \cdot \text{s}^{-1}$) through a plane at the beginning of the beam tube thimble, as calculated by the

INEL Monte Carlo calculations, is used as a measure of the source strength. For example, in going from the full overlap three-element core, with a 200- × 100- × 14-mm thick thimble, surrounded by the 3-mm wall of a water jacket, to a 100-mm-diam circular thimble which is much thinner and does not need a water jacket, the neutron current density increased by as much as 50%. For all cases with samples of reasonable height analyzed to date, the smaller thimble is better; this is a good example of the results from ongoing efforts to reoptimize the entire system as more data become available.

3.3.2 Isotope Production

An evaluation of the isotope production capability of ANS compared to HFIR was completed and published (Mirzadeh 1992). Estimated ANS and HFIR production yields of several medical radioisotopes of current importance are compared. These radionuclides are currently in clinical trials and include copper-67, tin-117m, dysprosium-166 (parent of

holmium-166), platinum-195m, and tungsten-188 (parent of rhenium-188). A discussion of the potential reactor production of molybdenum-99 by neutron capture of molybdenum-98 is also included in the Mirzadeh report. Molybdenum-99 is the parent of technetium-99m, which is the most widely used diagnostic radioisotope in nuclear medicine. A summary of the potential applications of these radioisotopes is given in Table 3.3.

For each of these isotopes, the specific activity (mCi/mg of target) vs time was calculated for ANS and HFIR irradiation facilities. For ANS, these were based on the conceptual design and unperturbed flux at the irradiation location. One example is shown in Fig. 3.10 for tungsten-188 where production in two of the ANS hydraulic tubes and the HFIR hydraulic tube is compared.

Table 3.4 summarizes the results for the isotopes considered. In general, the ANS radioisotope production capabilities will be equal to or better than those of HFIR. The capabilities of these facilities will offer an increased efficiency of radioisotope production, a

Table 3.3. Examples of reactor-produced radioisotopes currently under clinical trials^a

Radionuclide	Half-life	Example of applications and comments
Copper-67	61.9 h	Labeled antibody for soft tissue tumor therapy.
Technetium-99m	6.0 h	Labeled drugs and particles for various diagnostic applications. High specific activity is produced via the ⁹⁹ Mo/ ^{99m} Tc generator system.
Tin-117m	13.6 d	^{117m} Sn-DTPA ^b as palliative agent for painful skeletal metastases.
Holmium-166	26.4 h	Labeled antibodies for tumor therapy and as labeled particles for treatment of rheumatoid arthritis and cancer of liver. High specific activity is produced through decay of ¹⁶⁶ Dy/ ¹⁶⁶ Ho generator system.
Platinum-195m	4.0 d	Pharmacokinetics of antitumor drugs (e.g. "cis-platinum").
Rhenium-188	17.0 h	As labeled antibody for tumor therapy and as Re-DMSA ^c agent for treatment of medullary thyroid carcinoma and bone metastases. High specific activity is produced via ¹⁸⁸ W/ ¹⁸⁸ Re generator system.

^aAlthough widely used clinically, technetium-99m is included in this table for comparison.

^bDTPA = diethylenetriaminepentaacetic acid.

^cDMSA = dimercaptosuccinic acid.

Fig. 3.10. Production of W-188.

greater availability, optimized production, and, as a result, conservation and efficient use of rare or expensive isotopes used in production.

3.3.3 Positron Facility

The copper-64 positron source design remains essentially unchanged from last year's report. Copper-coated microspheres irradiated

in the reactor will be sent to the positron spectroscopy facility (PSF) for introduction into the source chamber. After the beam is generated and conditioned in the source, it will be transported to various detectors. An improved understanding of the hydraulic flight tube and rabbit design with associated thermal-hydraulics has been achieved.

The PSF conceptual design has been improved by the development of a more flexible, capable, user-friendly design. Substantial progress was made in developing the facility requirements, many of which are driven by the detector designs, especially data acquisition and resolution goals. It became apparent that these goals could best be met by locating the source and the detectors outside the reactor building. Hence, the PSF conceptual design was revised so that the source, detectors, and several associated support functions are located in a separate building on the ANS site. As shown in Fig. 3.11, rabbits will be transported between the PSF hot cell and the reactor building hot cell via a pneumatic flight tube. At the reactor containment building hot cell, they will be transferred

Fig. 3.11. Positron spectroscopy facility schematic.

Table 3.4. A comparison between the Advanced Neutron Source and the High Flux Isotope Reactor for production of copper-67, californium-252, molybdenum-99, tin-117m, dysprosium-166, platinum-195m and tungsten-188

Product radionuclide	Irradiation period	Production yield (mCi/mg of target)		Comments
Copper-67	8 d	ANS-TTF ^a	ANS-HT2 ^b	Approximate saturation yield
		2.1×10^{-1}	8.0×10^{-1}	
Molybdenum-99 Tin-117m	11 d 17 h	ANS-HT4	ANS-HT2	Approximate saturation yield
			HFIR-HT(#5)	
Dysprosium-166	16.3 h	2.1×10^2	2.8×10^2	Approximate saturation yield
		9.2	6.3	
Tungsten-188	17 d	1.3×10^4	9.4×10^3	Maximum yield
		1.3×10^2	1.6×10^2	
Platinum-195m	1.2 d	2.1	3.6	Saturation yield
			2.5	
Californium-252	1 year	ANS-TTF	HFIR-TTF	
		1.5 mg	~1 mg	

^aTTF = transuranium target facility.

^bHT = hydraulic tube.

from the pneumatic tube into a hydraulic tube that transports them to the irradiation station located in the reflector vessel. After irradiation, rabbits will be returned to the PSF hot cell by the reverse route.

The positron spectroscopy building will be sited in the area set aside in the conceptual design for future expansion. It will provide space for up to four positron sources, a full complement of detectors, and essential support functions. Detectors are expected to include defect profiling devices, a large angular correlation of annihilation radiation (ACAR) camera array to be used for electronic structure determination, surface analysis devices, and other positron spectroscopy techniques. Laboratory space requirements are dominated by the 50-m-diam $\times$ 45° ACAR detector depicted in Fig. 3.12. The facility should be sufficiently flexible to accommodate continued user community growth and new research techniques.

Position resolution goals require that the ACAR detectors should be dynamically

stable in a relative reference frame to within approximately ± 0.1 mm for periods of several minutes. For the positron annihilation microprobe to operate at its 0.1- μ m ultimate resolution, ≤ 0.1 -mm dynamic stability between the primary source and the demagnifying stages of the microprobe may be required over a period of several hours.

The positron building layout will segment the various functions. For example, laboratories and offices will be readily accessible, access to mechanical equipment will be appropriately restricted, and access to the radioactive sources will be very restricted. Other miscellaneous requirements include low gamma and radio frequency noise background for certain detectors, and minimum restrictions on experimenter access to the facility and on the types of samples that can be introduced into the facility. The PSF will rely on the main ANS complex for support services, including security, fabrication shops, computing, library, cafeteria, offices and conference rooms.

Fig. 3.12. Angular correlation of annihilation radiation detector layout plan view (in positron spectroscopy building).

3.3.4 Hot Source

The basic design of the hot source remains the same: an insulated 275-mm-diam graphite sphere heated by gamma and, to a lesser extent, neutron radiation to a temperature ≥ 2400 K and contained within a double-walled, light-water-cooled zirconium alloy vessel. This vessel is enclosed within a thimble assembly in the reflector pool in the same manner as the cold source.

Thermal performance of the hot source has been analyzed based on the best available heating rate and material property data. Beginning-of-fuel-cycle material heating rates were obtained from 3-D Monte Carlo calculations performed at INEL. A CBCF insulator developed in an ORNL M&C laboratory appears to be the most effective insulation material. Thermal conductivity measurements of CBCF and other insulation materials have been made, but conclusive results are not available yet.

A 3-D finite element model was developed to evaluate the effects of nonuniform heating, temperature-dependent property data, and large velocity changes in the annular cooling path. Results from the finite element model were verified by a nonlinear analytic conduction solution. CBCF thermal conductivity data was based on preliminary measurements obtained by the R&D program for temperatures up to 2300 K with analytic extrapolations to 2600 K. Further R&D work is needed to obtain measured conductivity data up to 2600 K, to evaluate the high temperature effects of sublimation and diffusion mass transfer, and to select the optimum heat treatment for the insulation material.

The midplane volumetric heating rate in the hot source as a function of distance from core centerline is shown in Fig. 3.13. Heating peaks in the metal containment shells and again in the moderator, with lower rates in the insulation proportional to the density. The temperature profile is superimposed, which indicates that a peak temperature of 2600 K is possible. Despite an 85% increase in the heating rate from the back to the front of the moderator, the moderator temperature profile is nearly flat because of the high conductivity relative to the insulation.

Cooling calculations were based on a water flow rate of 4.0 kg/s, which is sufficient to ensure turbulent flow (Reynolds number ≥ 10000) throughout the cooling annulus and to remove the 49.5 kW of heat generated in the assembly. Cooling velocities and heat transfer coefficients were calculated around the sphere, and wall temperatures at the convective interface were shown to remain below 65°C.

The use of heavy water as the hot source coolant is under consideration. Based on 3-D Monte Carlo calculations from INEL, attenuation of the available hot neutron flux by heavy water is 15–20% less than by light water. A heavy water cooling loop for the hot source would require only a 50–75 L increase in the ANS heavy water inventory.

Preliminary neutron physics analyses indicate that a hot source cavity (similar to the cold source cavity) may yield a significant increase in the hot neutron gain sector. Cavity optimization studies are planned for early FY 1995.

3.4 BALANCE OF PLANT

3.4.1 Combined Electrolysis and Catalytic Exchange Demonstration

The purpose of the CECE demonstration is to test certain key components of the ANS heavy water upgrade and detritiation facility (HWUDF) at close to actual operating conditions. These components are the electrolysis cell, the catalyst, the trickle bed recombiners, and the CECE column control scheme. Since the HWUDF is on the critical path for reactor start-up, failure to identify potential weaknesses in the CECE system could cause serious schedule delays during the heavy water processing period before initial reactor fill. The objectives of this demonstration program are as follows:

- (1) test Electrolyser Corporation's low inventory electrolysis cell at HWUDF operating conditions,
- (2) test the CECE column control strategy,
- (3) test the CECE catalyst with ORNL process water,
- (4) test the trickle bed reactor, and
- (5) develop an experience base with ORNL

Fig. 3.13. Temperature and heating rate profiles for the Advanced Neutron Source hot source.

personnel in the operation of this type of equipment.

The only electrolytic cell built by the vendor to date using this design was a prototype to be scaled up at a later time. The test electrolytic cell, which is designed to have a limited electrolyte volume, will be a full-size HWUDF cell. Oxygen from the cell is vented to the environment. Deuterium from the cell is passed through the catalyst columns. A compressor is not needed because the deuterium is produced by the electrolyte cell at a high enough pressure to operate the column. The catalyst columns are equivalent in length to one full-size HWUDF column but will be only approximately 80 mm in diameter, scaled to duplicate the liquid and gas fluxes in the HWUDF columns. Demineralized light water is fed through the top of the column. Deuterium in the gas from the electrolytic cells is exchanged with protium in this light

water stream, and the protium is vented. To simulate product withdrawal to the cryogenic distillation system from the electrolytic cells, a stream of gaseous deuterium will be removed from the stream flowing to the catalyst column, recombined, diluted with the addition of light water, and then fed back into the column as a feed stream. Removal of the gaseous stream in this manner is required for control system testing and will also serve to simulate a feed stream to the column, as well as to test the trickle bed recombiner technology. This recombiner will be appropriately sized for the stream flow rate. Heavy water is available at the Y-12 facility in the form of "half heavy water" (50% D₂O, 50% H₂O). This material will cost about one-third that of reactor grade heavy water and is preferred since the column is not long enough to maintain the full concentration gradient. The tritium content of this water is

low, and tritium spiking may be required in some tests, but the quantities used will pose minimal safety concerns. The deuterium concentration of the hydrogen vented from the column top will be natural background, so no loss of the material will occur. The equipment will be installed in an existing ORNL facility yet to be determined. The necessary dc power supplies are available at Y-12 in the ORNL Fusion Energy Division. A ventilation system will be required to ventilate the equipment where it is installed, along with a room hydrogen monitor. In addition utilities, such as cooling water, electrical power, and ventilation, will be required, along with the necessary permits, safety reviews, and environmental documentation.

To demonstrate the design, the electrolysis cell will be operated in a batch mode over about 1 year. Data on power consumption, throughput, efficiency, leakage occurrences, reliability, and ease of maintenance will be generated and used to refine the design of the actual cell for the facility. The CECE column computer model predicts that precise feed and withdrawal rates are critical to maintain low column effluent discharges and product isotopic purity. Control techniques will be verified by controlling the gaseous deuterium removed from the electrolytic cells, which will enable determination of the sensitivity of the system to load and/or set point changes and will be used to verify the computer model calculations. The trickle bed recombiner will be tested to demonstrate the design, efficiency, and reliability as well as to identify any problem areas. The catalyst will be tested to determine if there is any component present in ORNL water that could impair catalyst function (none are known at present). Although extensive catalyst testing has been performed, the operation of the catalyst is critical to the success of the plant, and the identification of any special water treatment requirements is warranted. Perhaps the most important goal is to develop an experience base at ORNL in the operation of this type of equipment.

The facility will be maintained in a standby mode at the conclusion of testing and will be available to simulate problems that might occur

in the full scale HWUDF after start-up and for training the actual facility operators. The facility could also provide very limited backup upgrading capacity in the event of an unplanned shutdown of the HWUDF. At the conclusion of the program, the cell will serve as a spare for the HWUDF. The catalyst will be retained as spare material, and the rest of the equipment will be removed for disposal. The heavy water will be returned to Y-12 inventories.

3.4.2 Primary Cooling System Studies

A study was performed to estimate the heat rejected to the various pools, in all operating modes and conditions, by the reactor primary coolant system (RPCS) piping and heat exchangers. This information is necessary to define system interfaces for the light water pool systems and to assess the contribution of, and the need for, the emergency heat exchangers during loss of the reactor secondary cooling system (RSCS). The study simulated the reactor primary cooling circuit, including all relevant components, and performed analyses to evaluate the heat rejected to the pools for different operating conditions and design-basis event scenarios. The configuration of the RPCS piping, the volumes of various pools, and the heat transfer correlations for the piping configurations shown in the CDR are included in ANSDM.

This study concluded that the ANS conceptual design can survive all the loss of secondary coolant events. It also concluded that, if all four of the heat exchanger pools are available and connected to the pipe chase, there is sufficient piping surface area to remove the decay heat without the emergency heat exchanger. However, connecting the pools would make the system susceptible to single failure. This study is published as ORNL/TM-12677 (Ibn-Khayat et al. 1994).

Documentation packages were prepared for the RPCS as an example for the development of the architecture and requirements for the ANS control computers software design documentation. The purpose of software

design documentation is to define and communicate requirements for software code and operating/maintenance procedures in a manner that results in humans and control systems working together to provide consistent and acceptable plant operation. To serve this purpose, the documentation must serve as a communication tool between process/mechanical system designers, software developers, operations personnel, and other design team members as well as define requirements completely, consistently, and correctly.

The three major components in the software design architecture and their relationship to other project documentation are shown in Fig. 3.14.

The ANS plant state diagram describes the major plant states and the transitions between them. It also provides the means to integrate the requirements for the operation of each individual plant system.

The system correlation data base defines the relationship between the plant states and the states of each individual system. This data base will provide the traceability from plant states to all system design requirements [via reference of system design description (SDD) sections], software design documentation (via identification of system states), and operations procedures (via identification of system states and process operation documentation) needed to support the operation of that plant state.

Software design documentation packages are required for each individual process or mechanical system. These packages define the possible states that can exist in each system and the human and/or control system actions that are required to transfer from state to state. These design documentation packages include:

- system overviews as described in Sects. 3–6 of the SDD;
- a state diagram showing all states applicable to the system;

- a simplified system diagram that shows key operating parameters;
- a state table that contains the operating parameters and set points for each instrument and piece of equipment;
- process operating documentation that shows the logical flow of actions to be taken by humans, equipment, and interfacing systems during a transition between system states; and
- software design notes that provide the logic and justification behind the requirements listed in the software documentation.

3.4.3 Seismic Design Spectra for ANS Structures/Systems/Components

The current reference site for ANS is on the ORNL reservation adjacent to HFIR. This site is located within what is commonly labeled the East Tennessee Seismic Zone. Figure 3.15 presents both historical and recorded seismic activity in the southeastern region of the United States.

DOE Order 5480.30 mandates that the design of ANS meet the requirements, including natural phenomena, that would be expected of a commercial reactor licensed by NRC. Those requirements for natural phenomena are promulgated in 10CFR50 and 10CFR100. Figure 3.16 presents conceptual design spectra that meet NRC/DOE requirements and envelopes the maximum expected ground shaking at any location within this seismic zone except New Madrid, Missouri, and Charleston, South Carolina. Additionally, Fig. 3.16 provides a comparison of time history response spectra plotted against the smooth design spectra. These time histories fulfill the pre-1989 NRC Standard Review Plan Sect. 3.7.2 requirements.

Some preliminary analytical seismic analyses have been performed for the ANSR building using the conceptual structural configuration and the time histories of acceleration scaled

Fig. 3.14. Software design documentation architecture.

Fig. 3.15. Historical and recorded seismic activity in the southeastern region of the United States.

to envelope the current horizontal and vertical design spectra. Table 3.5 presents the calculated dynamic characteristics of the reactor building complex. Figure 3.17 presents typical amplified earthquake motion at the reactor building operating floor for a damping value of 5%. These spectra may be used for the conceptual design of systems, components, and equipment.

Early in the Title I design process, more site-specific hazard curves will be generated using evolving NRC requirements. Additionally, time histories of acceleration will be created to meet the current NRC power spectral density requirements for frequency content.

3.4.4 Project Information Network

The project information network (PIN) will connect approximately 500 project participants

so that data can be shared and integrated. The network is shown in Fig. 3.18 and will be a key element in creating the tens of thousands of construction drawings, specifications, instructions, procedures, and other documents needed to support design, construction, and start-up. This will result in the generation and management of the equivalent of more than 100 million pages of paper during the 10-year design and construction phase of the project. Storage of this information as actual paper would require inclusion of very substantial file storage capacity as part of the facility, in addition to the labor costs associated with handling paper. Further, since the ANS Project will implement all of the functions needed for any major engineering/construction project, the techniques and software developed for this network can be readily used by other engineering/construction organizations to support major construction projects.

Fig. 3.16. U.S. Nuclear Regulation Commission Regulatory Guide 1.60 (a) horizontal and (b) vertical shape spectra.

**Table 3.5. Advanced Neutron Source reactor building
dynamic characteristics**

Mode number	Frequency (Hz)	Participation factor (% mass)			Remarks
		N-S	E-W	Vertical	
1	7.50	22.98	68.84	0	Fundamental E-W mode
2	7.52	68.82	22.94	0	Fundamental N-S mode
3	16.53	0.87	2.90	0	Secondary E-W mode
4	16.59	3.12	0.96	1.67	Secondary N-S mode
5	18.38	0	0	95.42	Fundamental vertical mode
6	20.21	0.53	1.02	0	
7	20.35	1.56	0.88	0.86	
8	20.75	0.49	1.02	0	
9	22.33	0.50	0.83	0	
10	22.40	1.12	0.58	0	
11	34.73	0	0	0	
12	37.37	0	0	0.74	Secondary vertical mode
13	38.83	0	0	0.36	
14	56.14	0	0	0	
15	59.57	0	0	0.81	

The design/construction/start-up effort will include operations and maintenance personnel and will involve the generation of all operations and maintenance procedures, training materials, instructions, manuals, and other materials needed for operations at start-up. Therefore, by storing the data in electronic forms that are readily usable by operations, maintenance, and facility business computing system applications, the PIN will greatly simplify the traditionally expensive turnover activity of preparing data for operational use.

Thus, the main purposes of the PIN are to:

- support efficient production, generation, review, and approval of design and construction data by the project team and
- automate the turnover of this information to operations and maintenance personnel who will operate the plant.

Research performed for the Electric Power Research Institute (EPRI 1987), and recent experience in using 3-D computer models and

other computer-aided engineering techniques (Brake 1990, Warren 1994) shows potential savings of tens of millions of dollars. Analyses performed for ANS indicate that a \$5M investment in the PIN could produce a \$60M return in reduced costs.

Design, procurement, installation, start-up, and user training of the project office portion of the PIN have been completed and are beginning to be used for production purposes. A functional specification defining network design, implementation, and operation tasks for industry team and MMES activities was completed, and numerous data integration, software application requirements development, and software system pilot efforts are under way. The network presently serves more than 100 users, stores more than 1 gigabyte (approximately 300,000 pages) of data, and is one of the largest networks at MMES.

In this area, the most significant task accomplished this year was to determine how to integrate the ANS human activities effectively into the technology offered by the PIN. This will

Fig. 3.17. Typical amplified earthquake motion at the reactor building operating floor for a damping value of 5%.

be accomplished by using a graded approach to the use of paper, standard formats for work produced, software support, and compliance with Continuous Acquisition and Life-Cycle Support (CALS) standards. This graded approach divides all user-generated data and documentation into working files and official project files. Working files are documents and/or data that are in the process of being generated but that have not been officially submitted to the project. These files exist in a combination of electronic and paper formats and may reside on user workstations, on

PIN servers, or in user offices. When working files are officially presented to the project for review, comment, approval, or other project activities, they become transmitted files. These files are expected to be mostly in electronic form, which will reside only on PIN servers (or, if not in electronic formats, in official project paper file storage systems). Working files become transmitted files via an electronic transmittal function that collects all information pertinent to the document or data being transmitted so that the data can be properly managed.

Fig. 3.18. Advanced Neutron Source Project information network.

Further, the way that users perform their work, the list of personnel needing to view and review that work, configuration management functions, the use of the PIN, and nearly all other functions will change when working files are transmitted to the project as official data, so the requirements will be different for these transmitted files. When users are generating working files, the PIN functions primarily as a user tool that provides software services and data. When handling transmitted files, the PIN becomes primarily a project tool that provides appropriate configuration management, data protection, and storage. This approach is shown in Fig. 3.19.

Transmitted files need to be viewed by many project team members and must have appropriate storage, protection, tracking, and other management provisions. Requirements focus on allowing all users to view and review the data and ensuring that appropriate configuration management activities, data protection and

Fig. 3.19. Graded approach to network use.

storage, and other project functions can be performed. Users cannot change transmitted files. Generating the next revision of a document or data by changing the latest revision is performed by furnishing the users an electronic copy that they may use as a working file.

3.4.5 Design Concept for Application-Specific Integrated Circuit-Based Reactor Protection System

Application-specific integrated circuits (ASICs) are being developed to demonstrate their capabilities and potential advantages over computers for use in nuclear safety systems. These circuits may be analog or digital or a hybrid of the two, but the ones being developed for this task are digital circuits. This work is being done as part of a cooperative R&D agreement (CRADA) between the ANS program and EPRI. This agreement came about because of licensing problems in the nuclear industry with the use of computers and software in safety systems and because ANS has a need for diversity among redundant protection systems and requires a faster speed of response than currently available in qualified, commercial nuclear protection systems. Potential advantages of ASICs over software-based safety systems are that ASICs can be tailored to perform specific functions with little overhead; the ASICs design is diverse in that it has different failure modes from software; verification and validation of ASICs may be simpler and less expensive than for software; ASICs may reduce common-cause failure probability by implementing independent functions in separate hardware.

The capabilities of ASICs are demonstrated by designing circuits to implement typical safety system functions. This demonstration implements safety functions of a typical pressurized-water reactor (PWR) pressure channel and delta-T/Tavg (DTTA) channel. The pressure channel is relatively simple, and the DTTA channel is the most complex one in a PWR. The pressure channel is typical of most protection system channels in a PWR and in the ANS. The DTTA channel is of comparable complexity to the neutron flux signal conditioning in the ANS protection system. One algorithm for the DTTA channel is shown in Eq. 3.

$$\Delta T_{odt} = \Delta T_0 \left[K_1 - K_2 \left(\frac{1 + \tau_1 s}{1 + \tau_2 s} \right) (T_{avg} - T_0) + K_3(P - P_0) - f_1(\Delta I) \right], \quad (3)$$

where

ΔT_{odt} = calculated set point;

$\Delta T = T_H - T_c$, where T_H and T_c are the calculated hot and cold leg temperatures, respectively;

K_x ; τ_y ; T_0 , and P_0 = constants;

s = the Laplace transform variable;

$T_{avg} = \frac{T_H + T_c}{2}$;

$f_1(\Delta I)$ = the flux difference modified by a nonlinear function;

P = power.

The concept for implementation of safety system ASICs is to have flexible designs that can be used in several applications by plugging in a new controller. Flexible design is achieved with basic modules that can be combined in different ways to perform many higher level functions. The basic modules include a comparator, multiplier, adder, and square-rooter that can be combined to implement functions such as comparators; lag, lead/lag, and rate filters; nonlinear input/output conversions; quadratic solvers; and difference and average value computations as shown in Fig. 3.20.

Two algorithms similar to the one in Eq. 3, along with some signal conditioning, are completed in less than 2000 clock cycles—2 ms for the demonstrator, for which the clock oscillator operates at 1 MHz. After completion of the safety algorithms, the controller transmits selected parameters via a digital input/output (DIO) interface to a computer for display. Constants and set points for the demonstrator are stored in erasable, programmable, read-only memory (EPROM) and read back for display and confirmation on the computer. Computations are done using fixed-point arithmetic. For accuracy

Fig. 3.20. Block diagram showing basic modules used in the application-specific-integrated circuit-based reactor protection system demonstration.

and range, this requires a 42-bit bus between basic modules. Random-access memory stores intermediate results and values from previous time steps for recursive computations used by the filters. The controller is designed such that it cannot lockup in an undefined state. If an error occurs anywhere in the system, the safety function will not fail because the controller always returns to start under control of a counter.

A comparator output can trip high, trip low, latch or not latch, and it can include hysteresis to prevent output chatter. The trip output is failsafe for loss of voltage or system clock. This is done by outputting a string of clock pulses when it is untripped such that clock failure generates a trip. Trips that should not actuate on loss of clock can be enabled through a process opposite to this one.

SAFETY 4

Efforts to identify safety-significant requirements and phenomena were intensified this year. The basic objective is to ensure an expeditious safety certification process by achieving significant progress toward closure of all potential safety issues **before** the submittal of the preliminary safety analysis report (PSAR). The following areas have been identified for concentrated attention: pressure boundary integrity as it relates to the definition of design-basis failure scenarios for accident analysis; the potential for fuel damage as a result of core flow blockage; and severe accident-related phenomena that could, at least hypothetically, affect containment integrity. A diverse approach to closure is being followed in each area, including external peer reviews, increased integration between safety analysts and designers, improvement of analytical capabilities, and enhancement of the basic physical data base.

Major achievements of the year include holding external peer reviews; working on the ANS severe accident plan; updating the conceptual safety analysis report (CSAR) loss of off-site power (LOSP) and pressure boundary failure accident analyses; completing an ANS-specific core flow blockage risk assessment; developing of CFD methods for analyzing core flow blockage and initial benchmarking of the CFD models against actual flow blockage experimental data obtained at ORNL; completing of certain severe accident analyses aimed at issue resolution; and defining specification of design requirements in the PDR to reduce the risk of

core flow blockage and to justify the exclusion of major pressure boundary failure from the design basis of the reactor. Each of these areas is described in some detail in the following sections.

4.1 SAFETY DESIGN CRITERIA

The PDR defines the plant-level requirements for ANS and controls the plant's baseline configuration. A vital role of the PDR is to communicate important nuclear safety requirements to designers. Progress this year (revision 5) includes the reorganization and revision of PDR Sect. 4, "Safety and Certification Goals and Requirements," to facilitate the flow of requirements into the design process and to define specific requirements that support identified risk reduction initiatives. The changes clarify safety goals and requirements, equipment safety classification levels, implementation of recent DOE regulations and directives, NRC regulations and guides, and other guidance documents. The result is a more concise and easily understood set of fundamental requirements for safety and regulatory compliance that addresses both new and revised DOE requirements, as well as comments received during the DOE review in December 1992. Of primary importance is the recent DOE rulemaking to implement the Price-Anderson Amendments Act and the recent DOE 5480 series of directives that cover the environmental

protection, safety, and health protection programs for DOE operations.

Revision 5 adds new sections to the PDR to focus project attention on two major risk reduction initiatives: core flow blockage and large pipe break. The ANS Phase 1, Level I probabilistic risk assessment (PRA) has concluded that the risk of core damage for ANS is dominated by core flow blockage (see Sect. 4.3). The new PDR section on core flow blockage requires the evaluation of design features and the implementation of institutional controls to prevent and mitigate core flow blockage.

Consistent with the New Production-Heavy Water Reactor (NP-HWR), the Fast Flux Test Facility (FFTF), and the Clinch River Breeder Reactor Plant (CRBRP), revision 5 of the PDR defines the requirement for elimination of the double-ended guillotine break (DEGB) from the design basis of ANS. Nine criteria, consistent with those recently adopted for stainless steel piping of the NP-HWR and similar to those imposed in the past by NRC for FFTF and CRBRP, provide the basis for eliminating the DEGB for stainless steel piping. An additional criterion (i.e., beyond the NP-HWR criteria) on use of the ASME operations and maintenance manual, Part 3, for vibration was added. Work is currently under way to demonstrate a basis for eliminating the DEGB for aluminum components.

The discussion of specific instrumentation requirements for primary coolant leakage detection received considerable attention during the project working group meetings for revision 5. The general requirement selected is that leakage shall be detected in sufficient time to allow reactor shutdown before a rupture occurs of a size sufficient to cause fuel damage. This requirement considers (1) the leak detection sensitivity achievable at the Savannah River heavy water reactors for detecting tritium in air compared to the capabilities for detection of heavy water leakage into light water (2) the operability and fragility considerations for heavy water leakage detection systems, and (3) the much less stringent licensing requirements for the CRBRP. For the CRBRP, the NRC accepted

a leakage detection sensitivity comparable to that which they accept for licensing light water reactors as sufficiently sensitive for eliminating the DEGB from the CRBRP design basis.

The goal for limiting worker risk resulting from radiological accidents has been improved in revision 5, bringing it into closer correspondence with new DOE guidelines. (The goal is now specified at 10% of the radiological risk associated with normal occupational radiological exposure on the site.) Furthermore, the worker risk goal is now designated to apply to the maximally exposed group of workers on the ANS site. In a similar vein, the radiation exposure acceptance criteria for the analysis of design-basis events, from anticipated to extremely unlikely, have been expanded and updated. A new table provides specific worker radiation exposure limitations, and the exposure limits for off-site residents have been more completely specified.

Core melt frequency is useful as a nuclear safety index to assess the overall plant nuclear safety or the impact of a proposed design change. Revision 5 of the PDR leaves in place the existing goal for limiting the mean value of severe core damage frequency to $\leq 10^{-5}$ /year for internal events, but now states an additional, related requirement that the mean value of severe core damage frequency shall not exceed 10^{-4} /year, including internal events and other events, such as fire or flooding. The new requirement removes a potential ambiguity by providing **both** a goal and a requirement. To ensure that the goal (10^{-5} /year) receives adequate consideration in the design process, a provision is added for the cost-benefit analysis of radiological accident-related risk reduction measures. A viable cost-benefit evaluation would include such issues as a change in risk to the public or to the workers because of the design change; the cost addition or reduction because of the proposed design change; a potential safety impact on facility or operational complexity because of the design change; and the practicality of the design change with respect to the overall facility design, operational missions, or other objectives. Most importantly, the evaluation

must consider whether a proposed design change decreases the hypothetical severe accident risk only at the expense of increased routine radiation exposures.

4.2 PRESSURE BOUNDARY INTEGRITY

Work was initiated in 1994 to develop a case for elimination of instantaneous DEGB from the design basis of ANSR primary coolant piping. Preliminary analyses were completed to quantify estimates of operating and accident loading conditions on ANSR piping and other primary pressure boundary components. The analyses included estimates of seismic loads that would be imparted to primary pressure boundary components. The development of preliminary loads and stresses was completed for components fabricated using both stainless steel and aluminum.

A special ANS Pressure Boundary Integrity Senior Review Panel was established to review the project's analyses and to evaluate the potential for elimination of the instantaneous DEGB from the ANS design basis. This panel is essentially the same group that had endorsed eliminating the instantaneous DEGB from the design basis of DOE's NP-HWR (ORNL/NPR-92/64, January 1993). The panel consisted of experts from the ASME B&PV code committee and from the aluminum industry: Spencer H. Bush (Chairman), Robert J. Bosnak, Robert P. Kennedy, Everett C. Rodabaugh, Gerald C. Slagis, Gery M. Wilkowski, and Maurice Sharp.

The panel met in Oak Ridge March 1-2, 1994, and their final report was received at ORNL the following month. The review group agreed that, for stainless steel material, the instantaneous DEGB should be eliminated from the ANS design basis. In addition, the panel posed a set of questions that must be answered to extend elimination of the DEGB to aluminum components, saying:

Aluminum alloys such as 6061-T6 have not currently been shown to

meet the DEGB requirements based on information available to the Panel. Specific concerns and suggested actions to determine if aluminum will be exempt from the DEGB are given. Testing procedures and analyses covering non-destructive examination, leakage testing, and probability margins are given. The question of a simulated seismic test to determine whether aluminum can be exempt from the DEGB as a Design Basis is covered with details of a proposed program. The Panel believes substantial additional information must be collected, including simulated seismic tests, and fracture mechanics data before they can address positions on Aluminum alloy components.

A draft of the ORNL plan for aluminum components was issued for internal review. The plan provides for load-deflection testing of full size aluminum components to resolve the senior review panel's questions on fracture mechanics and seismic loading of aluminum components. It is planned to complete some of this testing in FY 1995. At an appropriate future date, the panel will be reconvened to review the progress of the project in answering those questions.

4.3 PROBABILISTIC RISK ASSESSMENT

The results of the Phase 1, Level I PRA of ANS show that, among all internal initiating events, core flow blockage dominates the risk of fuel damage. The Phase 1 (revision 0) results are qualitative because they were obtained by a design comparison with the HFIR data flow blockage probability assessment. A subsequent priority, addressed during the current year, was to address flow blockage with flow blockage analysis specific to the ANS baseline (two-element) design. In the effort to determine the probability of flow blockage, the following fundamental questions had to be answered.

1. What is available to cause a flow blockage event?
2. How likely is a failure to occur that would release debris into the primary coolant system?
3. Does released debris actually reach the fuel?
4. What is the critical flow blockage size that can be sustained by the fuel plates before fuel damage?

The first question was answered by asking each of the system engineers to identify the potential for generating debris from their system where such debris could migrate to the primary coolant system. Table 4.1 gives a sample of the data gathered. Concurrently, T/H analysis was under way (see Sect. 4.4) to determine, analytically and experimentally, the limiting blockage size that would cause fuel damage.

This size, believed to be approximately 10 mm for the conceptual design two-element core, considers the presence of a 50-mm unheated fuel plate entrance length. In order to account for uncertainties in this quantity and to introduce conservatism into the analysis, the event trees were analyzed assuming that the onset of fuel damage occurs for a blockage size of only 5 mm.

The second question required the elicitation of expert judgment as presented in Mosleh, Bier, and Apostolakis (1987) and Meyer and Booker (1990). Expert judgment is used to estimate the occurrence of an event where little or no directly applicable statistical data exist but where there is a solid knowledge base regarding both the design and expected operational practices. The responsible engineering designers of the RPCS, the reactor pressure and inventory control system, the reactor instrumentation and control system,

Table 4.1. Potential flow blockage debris

System	Debris	Source	Scenario ^a	Material	Installed barriers
Refueling	Screw, retaining rings, fasteners, etc.	Refueling tool head	L3	Stainless steel	None
I&C ^b	Debris remaining after control rod change-out	Tools, control rods, or other material	L1	Metallic	None
Primary	Check valve disk	Pump discharge check valves	L2	Metallic	RPCS ^c loop strainers
Pressure control	Check valve internals	Makeup pump discharge check valves	L3	Metallic	RPCS heat exchangers, pumps, strainers
Refueling	Small seal fragment	Irradiation capsule	L1	Nickel	RPCS loop strainers
Refueling	Insulation	Refueling tool head wires	L3/1 or L3/2	Ceramic	RPCS loop strainers, none
Reactor protection	Pressure taps and RTD ^d thermowells	Pressure taps and RTD thermowells	L1	Stainless Steel	RPCS loop strainers
I&C	Upper support springs, rollers, and roller axles	Inner control rods	L1	Inconel	RPCS heat exchangers, pumps, and strainers
Experiment	Transuranic rods	Transuranic assembly	L2	Aluminum	None; located directly under fuel

^aDebris originating between the core outlet and the loop strainers (L1), between the three operating loop strainers and the core outlet (L2), or from outside the primary pressure boundary (L3).

^bI&C = Instrumentation and Controls (Division).

^cRPCS = reactor primary coolant system.

^dRTD = resistance temperature detector.

the reactor power level control, the transuranic support assembly, and the refueling systems were the domain experts who provided input to estimate the flow blockage initiating event frequency. Based on their state of knowledge about the design of the systems, the components, or the assemblies, the experts were asked to bound the frequency of failures that could introduce debris into the primary system and to estimate the most likely size of that debris using the discrete ranges of less than 5 mm, 5–10 mm, or larger than 10 mm. On the average, 78% of the debris sources were estimated to exceed the limiting 5 mm size, with 45% expected to be larger than 10 mm.

Three scenarios are defined to categorize how and at what location the debris enters the primary system: L1 scenarios include debris originating between the core outlet and the loop strainers, exclusive of the strainers themselves; L2 scenarios include debris originating between the three operating loop strainers and the core inlet, including the strainer and (for the current two-element baseline) the transuranic support assembly; and L3 scenarios include debris that is introduced to the primary system from outside the primary pressure boundary. Contributors to L3 are refueling, maintenance, and adjacent systems, such as the pressure and inventory control system and the reflector vessel cooling system (RVCS).

The conceptual design of the transuranium assembly is a cylindrical arrangement of 30 target rods supported vertically and horizontally by retaining rings at the ends of the rods. A horizontal retaining ring in the center section of the rods provides added stability in a high flow regime. The knowledge that the transuranium assembly poses a risk to flow blockage has prompted efforts to relocate this assembly to a downstream position in the three-element core design discussed elsewhere in this report.

L3-type debris could enter the primary system during refueling operations, CPBT changeout, work on components within the reflector vessel, or during periods when the primary system is open for maintenance/inspection. The limited access to the

upper pressure boundary assembly (vessel head) during operation, refueling, and maintenance minimizes the types of debris that need to be considered. The most likely source of debris is from the refueling machine or from materials left behind after maintenance, but could also be material from the inner control rod assembly or transuranic support assembly, which are also removed during refueling. Depending on its point of origin, the debris would then follow the event sequence of L1 or L2 category material, designated, respectively, L3/1 and L3/2 subcategories. Maintenance items, for example, could include a glove, a badge, a piece of tape, a small wrench, or a small piece of wire. Strict inventory practices similar to those used at HFIR are assumed to be followed at ANS as called for in revision 5 of the PDR.

To answer the third question, the primary and interfacing systems were examined for existing debris filters. In L2 and L3/2 sequences, only the 50-mm unheated fuel plate region at the inlet of each fuel element allows flow reestablishment behind a core inlet blockage. The main defense against debris reaching the core, from the L1 and L3/1 scenario perspective, is the strainer in each primary loop. The primary loop strainers are constructed of heavy gauge stainless steel with basket perforations sized to strain out all particles originating between the core outlet and the strainer except those that could easily pass through the flow channels of a fuel element. The strainer failure mode of concern would occur if debris entered a strainer that has a local failure too small to allow detection of the drop in strainer pressure loss but large enough to allow a piece of material larger than 5 mm to pass on to the core. Each loop is equipped with its own strainer, making the likelihood of flow starvation resulting from strainer blockage as a precursor to fuel damage very low. Conservative estimates for strainer availability were taken from the PRA for HFIR and from the Savannah River Plant Generic Data Base.

Each scenario is modeled with an event tree that describes two conditional branch points that determine, given the initiating event from the expert elicitation process, whether existing

barriers are available to retain the debris and, if not, whether the debris is large enough to cause flow blockage that damages the fuel. The quantification process was carried out using Microsoft Excel[®] (Microsoft Corporation, Seattle, Washington) and Crystal Ball[®] (Decisioneering, Inc., Denver, Colorado), which calculate uncertainty distributions for each variable using Monte Carlo simulations.

The results of the initial quantification attribute over 90% of the risk of flow blockage to the L2 scenarios, specifically to the transuranic assembly because of its location upstream of the upper fuel element. The L1 and L3 scenarios' contribution to core flow blockage risk were more than two orders of magnitude below that of L2 because of the strainer in each loop, the limited access to the primary system, assumed maintenance practices comparable to or more stringent than those at other research reactors, and strainers in interfacing systems. These results indicate that efforts should focus on a robust transuranic assembly design that still meets scientific requirements (especially if it were to remain in a downstream location; see following paragraph) and designing the loop strainers to minimize their failure probability resulting from breakage, fatigue, or maintenance practices.

The proposed three-element core design has advantages related to the flow blockage risk. The transuranium assembly will be relocated to a position at the outlet of a fuel element, placing it in the L1 scenario rather than the L2. This means that any debris introduced by a failure of the transuranic assembly would likely be retained by one of the three operating loop strainers. Preliminary calculations indicate a reduction in the mean value of the flow blockage probability to below 10^{-5} /year. Another advantage, in this respect, to the three-element design concept is the possibility of greater tolerance in the critical debris size that could cause fuel damage; this further advantage is expected because of the lower power density and heat flux anticipated for the three-element core (and, of course, assuming that at least the 50-mm unheated length is included in a three-element core design).

4.4 TRANSIENT THERMAL-HYDRAULIC ANALYSIS

4.4.1 RELAP5 Code and Model Modifications

The primary system loop model was updated during this period, reflecting changes in the ANSR primary system design. The new model (Fig. 4.1) is based on an isometric model and features finer nodes, which are better able to capture acoustic wave propagation during rapid depressurization accident (RDA) simulations. In addition, the model includes tables of heat transfer coefficients vs temperature difference between the pipe outer wall and the pool for calculating decay heat removal by natural convection. It also includes updated elevations, piping length changes, accumulator and pressure sensor relocation, etc. Both main and emergency heat exchangers were remodeled and calibrated against Asea Brown Boveri (ABB) Combustion Engineering heat exchanger performance calculations. This calibration is necessary because the tortuous flow paths through varying and complicated cross-sectional areas on the shell side of the heat exchanger make flow conditions less certain there. Pressure drop through the RELAP heat exchanger model was matched to the ABB model by adjusting appropriate loss coefficients, while the shell side hydraulic diameter was adjusted until heat transfer predicted by RELAP and calculated by ABB Combustion Engineering agreed to within approximately 5%. Additionally, the main circulating pump performance curves were changed to a Westinghouse design that maintains a higher flow rate at a lower developed differential pressure.

A detailed model of the RSCS was also developed. It consists of four identical loops, of which three are active under normal operating conditions. Each loop removes one-third of the heat load carried by the RPCS. This heat is

Fig. 4.1. Nodalization diagram for the RELAP5 Advanced Neutron Source reactor system model.

then rejected to the ultimate sink by evaporation within the cooling towers. The mechanical draft of the cooling tower is represented by a controlled mass flow junction in the model. When the RSCS pump is operational and forced convection is taking place, the model is designed so that flow in the hot leg of the secondary loop proceeds from the primary heat exchanger to the hot leg riser. The flow then is assumed to proceed underground through concrete piping to the cooling tower, which is fixed at atmospheric pressure by a time-dependent volume. The coolant returns to the reactor heat sink cooling pool by means of underground piping (these are the cooling tower basins, and there is one per loop). This pool has a volume of 1680 m³, large enough to ensure an adequate inventory of cool light water if a power outage or similar incident were to disable the forced convection loop through the cooling tower. The surface of the pool is also fixed at atmospheric pressure. The secondary coolant pump draws water through a pipe from the bottom of the pool and is located at the highest point in the RSCS loop.

The RSCS and the RELAP5 model also incorporate capabilities to cool the primary system via a natural circulation loop that is able to remove sufficient heat to ensure safe condition after reactor shutdown. This loop is enabled when the pressure at the secondary side pump outlet drops below a critical value (assumed for testing purposes to be 200 kPa). As this low pressure reading is an indication of a rupture in the piping or a loss of pump head, the forced convection loop is disabled. To accomplish this objective, valving is reoriented to prevent backflow and to activate the natural convection loop. This loop draws cool water from the bottom of the pool, passes it through the heat exchanger, and deposits the heated water in the top of the pool. The pool's thermal inertia is so large that shutdown cooling under these conditions would be effective for many hours.

A detailed RELAP5 model of the RVCS has now been completed. The RVCS, like the primary and secondary systems, consists

of four identical loops. Three of them are comprehensively modeled, while the parts of the standby loop that are isolated by valves have been omitted for simplicity. The reflector vessel was assumed to have a volume of 40 m³ and to surround the CPBT. Under normal operating conditions, 17.5 MW of heat is assumed to be transferred by convection from the CPBT to the reflector vessel coolant. The model includes the hot leg riser, which distributes flow to the three active loops. Check valves are incorporated in each of the loops and are assumed to be open under normal conditions. Flow then passes through the pump and the heat exchanger. The letdown system is also modeled, with 1 kg/s of heavy water assumed to flow through the letdown/makeup system; its primary function in this model is to control the pressure of the system. In the model, this control is accomplished by fixing the pressure at the top of the letdown tank at 135 kPa by use of a time-dependent volume. The makeup from the letdown tank is then injected through a 38-mm- (1.5-in.-) diam pipe into the hot leg distribution area. Most of the flow leaving the main heat exchanger, however, enters the cold leg and returns to the bottom of the reflector tank.

Three thermal limit criteria in addition to the Costa OSV correlation have been considered in evaluating the safety margin for reactor transients simulated with the RELAP5 T/H model. The first is based on a modification of the S&Z correlation. Unlike Costa, the modified S&Z correlation has a reasonable low-subcooling limit; however, the uncertainty of this correlation at low velocities is very large. The second limit that has been added to the analysis is the $T_{wall} = T_{sat}$ criterion, which is exceeded if the fuel surface temperature as predicted by the Petukhov (1970) correlation exceeds the bulk saturation temperature in the coolant channel: this is an obviously conservative and very well known criterion that, if met, would lead to a very robust and easily documented safety argument. The third limit being considered is the Bergles-Rohsenow (1992) IB correlation.

4.4.2 RELAP5 Transient Analysis

4.4.2.1 Update of the Conceptual Safety Analysis Report

A series of calculations has been completed in order to update the CSAR to reflect recent changes in the reactor design. The major model changes were highlighted above. Three separate LOSP cases have been run, and a series of break cases have also been calculated to evaluate the response of the latest baseline design to core inlet and outlet breaks.

In the first and least severe LOSP examined, the primary loop pumps, the pressurizing pump, and the secondary loop pumps are all assumed to coast down immediately upon LOSP. In this case, it is assumed that ac power is recovered within 60 s. Credit was not taken for any automatic or manual insertion of control rods other than scram of the primary reactor shutdown rods: specifically, there was no control rod motion before the scram, no anticipatory scram, and the secondary shutdown system was not operated. Although designers are considering the specification of solenoid dump valves on the letdown valve air supply lines to force rapid closure of the letdown valves upon LOSP (they are spring-to-close, air-to-open), the letdown valves were conservatively assumed to remain open under normal automatic control after the LOSP in order to avoid a safety-related designation for the quick closure function that is under consideration. For the LOSP-1 case, the calculated core response must meet the anticipated event acceptance criteria, including no occurrence of departure from nucleate boiling in any hot spot and no occurrence of FE in a hot channel; both limits are being evaluated at a 99.9% nonexceedance probability level because loss of off-site power is an anticipated event.

At time zero, the primary coolant pumps, the secondary coolant pumps, and the pressurizing pumps all immediately began to coast down. The total core flow declines from the nominal value of about 2000 kg/s and coasts down to pony motor flow (10% of the nominal flow)

over approximately 100 s. After exceeding the flux-to-flow ratio limiting safety system setting at approximately 0.6 s, the reactor scrams, and the power drops sharply, approaching the decay heat level. The depressurization rate for this event is very slow, with pressure decreasing only as a result of coolant loss out of the letdown valves and the primary coolant pump coastdown.

Figure 4.2 compares FE minima (as predicted by the Costa correlation) and the CHF ratio, defined as the ratio of the calculated CHF or FE heat flux to the local heat flux, as predicted by the Gambill/Weatherhead correlation. Both minima are higher than unity, indicating that the predicted thermal limits are not exceeded for this event. The CHF was shown to be the most limiting. Note that the CHF is not predicted by RELAP5 if the fuel surface temperature is less than the local coolant saturation temperature because there can be no CHF, or even boiling, under those circumstances.

The second LOSP event, LOSP-2, includes the total failure of the primary scram system but successful operation of the secondary scram system after slight delay since the secondary scram is slightly slower than the primary scram. The objective of this calculation is to verify acceptable shutdown margin using only the secondary scram system. This is an extremely unlikely or a beyond-design-basis accident, therefore the safety margin was calculated against 95% nonexceedance probability CHF/FE limits. T/H results are similar to LOSP-1, with the thermal limit ratios characterized by two flow-driven thermal limit ratio minima. The first minimum, in this case, decreased below the LOSP-1 minimum, but the difference was insignificant because the difference in scram delay (30 vs 130 ms) was small.

LOSP-3 is the same as the base case (LOSP-1) but even more extreme in that the diesel engines do not start, off-site power is not recovered, and the primary system depressurizes totally before the pony motor batteries are exhausted. The calculations were extended up to 2500 s because the minimum design life of the batteries is 1800 s. LOSP-3 is categorized as an extremely unlikely event, hence, the safety

Fig. 4.2. Loss of off-site power-1 event flow excursion and critical heat flux thermal limit ratio.

margin was evaluated using 95% nonexceedance probability CHF/FE limits. The objective of this study is to verify transition to natural circulation under a total primary system depressurization. This calculation demonstrates that the passive cooling capability does not depend upon maintenance of primary system pressure and makes the LOSP-3 analysis conservatively applicable to the transition to long-term cooling after an RDA. In order to force the system to depressurize totally, it was necessary to open the three letdown valves to the full-open position gradually. The opening of these valves was an arbitrary further failure to increase the severity of the incident.

In the early part of the transient, the T/H behavior was much the same as the base case. Once the batteries were exhausted (at 1800 s), the pony motors began to coast down, the loop began to establish natural circulation, and the total core flow decreased to a minimum of

about 24 kg/s at the transition point to natural circulation. At this point, the temperature rise over the core reached its maximum, and natural circulation was initiated. The flow initially overshoot the steady-state value slightly and reached equilibrium several minutes later. The transition to natural circulation caused a second minimum in the FE ratio. However, even under depressurized conditions, the thermal limit ratio remained above 10.

Loss of coolant accidents were also reevaluated for the CSAR update. Each accident simulated involves an instantaneous rupture in the primary pressure boundary at either the core inlet or exit location. This rupture results in a depressurization of the system and loss of primary coolant (D_2O) to the light water pool. Although the accumulators inject D_2O into the primary system, they cannot respond quickly enough to mitigate the acoustic wave following the instantaneous break. A series of simulations

for instantaneous breaks at the core inlet and exit locations was run to determine the largest instantaneous break that could be sustained without violation of the relevant thermal limits. The design-basis 76-mm- (3-in.-) diam core inlet break was also simulated.

For each of the inlet breaks, a complex acoustic waveform originating at the break reaches the core exit before 0.1 s of the transient has expired. This is a result of transmissions and reflections of acoustic energy at various junctions and flow-area changes within the system. After about 0.1 s, the accumulator starts to affect the core outlet pressure. The system depressurizes down to the reactor pool pressure as the primary pumps coast down and the accumulators are drained.

The total core flow is affected by three factors: (1) the primary pump action, which tends to increase core flow; (2) the accumulator injection to compensate for lost break flow (the larger the pipe break, the larger effect this factor will have), which, especially for core inlet breaks, tends to decrease the core flow; and (3) the buoyancy forces and higher temperatures generated by the decay heat, which tend to increase the core flow. During the first 800 s of the transient, each of these factors diminishes, and the effect of decreasing accumulator flow competes with the effects of decreasing pump speed and decay heat. The total core flow begins to drop first because the pump is coasting down. Although the pump does not coast all the way down to the 10% pony motor speed until 100 s, the total core flow reaches a minimum and begins to increase again because of the declining accumulator flow, which is pushing less and less against the normal flow direction. Once the accumulator has emptied, the only one of the three core-flow factors that is still changing is the decay heat level; hence, the core flow begins to decline slightly with the reactor power level. If a simultaneous loss of off-site and on-site electrical power is assumed to occur, at 1800 s, into the transient the pony motor batteries expire, causing the primary coolant pumps to shut down altogether. After a short transition period from forced to natural circulation, the primary flow begins a slow, steady decline starting at 36 kg/s.

The 95% nonexceedance probability level thermal limits in the upper core are presented in Fig. 4.3 for the design-basis, 76-mm-diam, inlet break. As typical for an ANS RDA, three minima are observed in the FE thermal limit. The first is the result of depressurization following the break, the second is the result of decreasing core flow during pump coastdown, and the last, assuming simultaneous loss of all electrical power, is the result of decreased core flow after the pony motor trip during the transition from forced to natural circulation in the primary circuit. The thermal limits are not violated at any of these minima for this extremely aggravated transient. Larger instantaneous pipe breaks have also been simulated at the core inlet location. Although a 102-mm-diam break does not result in violation of either the FE or CHF thermal limit, the instantaneous 127-mm-diam break, combined with the additional failures, does lead to momentary violation of both FE and CHF.

In addition to the core inlet breaks, instantaneous core outlet (hot leg) breaks have been simulated to determine the limiting break sizes. Exit breaks behave similarly, with the depressurization wave originating from the break; however, the magnitude of the wave is smaller with the break at the exit location, rather than at the inlet, since the system pressure is lower at the exit. These calculations indicate that the largest instantaneous, circular orifice break that can be sustained at the core exit without exceeding the FE heat flux limit has a diameter of 127 mm (5 in.), while the largest break that can be sustained without violating the CHF limit has a diameter of 152 mm (6 in.).

4.4.2.2 Transient Design Calculations

An evaluation of the importance of pony motors during various transients was performed by parametrically varying battery lifetime. Both LOSP and RDA events were analyzed in this manner, with the RDA being the most severe. Both transients are the same as the LOSP-1 and RDA transients discussed in the previous section until the batteries expire, and the transition to

Fig. 4.3. A 76-mm design-basis instantaneous pipe break flow excursion and critical heat flux thermal limit ratio.

natural circulation begins. Therefore, only the behavior at this transition will be discussed here.

Figure 4.4 shows the FE thermal limit ratio for a 127-mm, instantaneous inlet pipe break (the limiting inlet break). In this case, the FE limit is exceeded during the depressurization phase of the event. However, the calculation was continued to show long-term safety margin behavior with and without batteries. For the case of no pony motors (or zero battery lifetime), the FE limit is violated because of the inability of natural circulation to prevent boiling in the depressurized core immediately after shutdown. The minimum FE limit ratio at the transition to natural circulation tends to increase as the battery lifetime increases. This increase results from the increased subcooling level and the lower decay heat levels with increasing battery lifetime.

The results show that the core can survive transition to natural circulation without exceeding

the FE limit for a nominal pony motor battery lifetime of only 30 min during both a station blackout and an instantaneous 127-mm-diam pipe break area near the core inlet. The results also show that, for the station blackout event, the FE limit is not exceeded, even without pony motors. In the pipe break event, however, the FE limit is exceeded if pony motors are not available. This is because a relatively higher pressure at the core exit can be maintained during the transition to natural circulation for the station blackout event than for the pipe break event. Therefore, it is concluded that pony motor batteries are needed and that the nominal battery lifetime ensures an appropriate safety margin.

Several other events have also been analyzed. A loss-of-flow accident was performed to study the effect on the safety margin during transition to natural circulation of reverse flow in one of the coolant loops. The event was initiated by

Fig. 4.4. Flow excursion thermal limit ratio at the upper core exit for various pony motor battery lifetimes during the pipe break event.

a main circulation pump (MCP) failure with a stuck-open check valve in one loop, while the other two loops operate normally. The two functional MCPs trip on a low pressure signal, but the battery-powered pony motors continued running for 30 min. Much of the flow from the two functional pumps would be diverted to loop 1 instead of the core because the discharge pressure decreases when the pump fails. Reverse flow in loop 1 occurs at about 2 s. Neither of the thermal limits ratios (CHF and FE) was exceeded during this event, even at 99.9% nonexceedance probability levels, since both the transition from reverse to forward flow in the failed loop and the transition to natural circulation conditions occur smoothly.

Pipe break analysis was also performed, assuming that the break occurs after the reactor

has scrammed and the primary pumps have coasted down to pony motor speed but while the system is still at high pressure. This scenario could happen during an earthquake occurring immediately after normal shutdown or as the result of an aftershock following an earthquake-induced shutdown. The break was assumed to occur instantaneously near the core inlet, but unlike the prescram break results, the acoustic wave is not the limiting phenomenon because the subcooling is so large that it is only slightly affected by the oscillations in saturation temperature. Such an accident, however, leads to diminished core flow during the depressurization and even to a flow reversal if the break area exceeds 0.008 m^2 (i.e., a 100-mm-diam break, which exceeds the design-basis maximum 76-mm-diam break size.)

This momentarily diminished flow occurs because the primary coolant pumps (running at 10% of full flow) are unable to maintain flow against the accumulator discharge during the initial part of the depressurization. The accumulators keep the core exit pressure high, but the core inlet pressure is low because of the reduced pressure at the break location; hence, the flow tends to reverse through the core. Parametric cases have shown that the worst time for such a pipe break to occur is approximately 90 s after reactor shutdown.

4.4.2.3 RELAP5 Verification and Validation

For the pipe break results obtained with RELAP5, the limiting phenomenon is the propagation of an acoustic expansion wave (low pressure) from the break site into the core region. Verification of RELAP5's capability to simulate pressure-wave transients has already been accomplished by comparisons with the method of characteristics. These comparisons showed significant dispersive and dissipative numerical errors, depending on the nodalization used. The limiting break size to exceed the FE limit has also been verified by a simple check using linear acoustic theory by considering the sound amplitude transmission and reflection coefficients at each abrupt area change. Six area changes that an acoustic wave would encounter moving from the primary coolant pump outlet to the fuel channel (in the downstream direction) have been considered.

Considering an instantaneous step change in the pressure of a volume just upstream of the primary system vessel adapter weld (PSVAW; i.e., an inlet break), the acoustic wave is first reflected off the narrowed sections of piping between the cold leg and the CPBT [called the inertial flow diode (IFD)]. This reflection occurs in the upstream direction and can be traced to the fuel channel by multiplying the appropriate transmission coefficients. Hence, the ratio between the amplitude of the rarefaction that reaches the fuel channel (by way of a reflection off the IFD) to the initial rarefaction

is 0.603. This calculation conservatively ignores the dampening effect of the break junction itself on the wave reflected from the IFD. Since the FE limit is exceeded in the 95% nonexceedance probability hot channel if the core exit pressure falls by more than 1.4 MPa (using the Costa correlation), the initial wave caused by the instantaneous break in the cold leg must have an amplitude of less than 2.32 MPa.

Table 4.2 gives the RELAP5-calculated initial amplitudes of the rarefaction waves for different break diameters. These were obtained from separate calculations using a single pipe component connected to appropriate upstream and downstream boundary conditions. Hence, this simple check confirms that a 102-mm-diam break at the PSVAW does not lead to violation of the FE thermal limit, but a 127-mm-diam break would.

Table 4.2. Calculated rarefaction wave amplitude caused by instantaneous breaks

Break diam (mm)	Pressure wave amplitude at the break site (MPa)
76	1.43
104	2.02
127	2.43
152	2.66

A similar, although not so definitive, analysis of the hot leg break has also been performed, and it concludes that RELAP is predicting reasonable limiting break sizes for this location as well.

The dynamic capability of RELAP to predict the FE phenomenon has also been evaluated (i.e., predicting the process as opposed to using a correlation to predict the onset of FE or OSV). Such predictions could be important when analyzing instantaneous break transients. In these transients, the most limiting conditions occur during the initial portions of the transient as the rarefaction wave propagates to the core exit and the local pressure approaches saturation.

This causes the thermal limit as predicted by correlations, such as Costa, to be exceeded since they are typically highly dependent on local coolant subcooling. The pressure depression lasts only a few milliseconds (see Fig. 4.4, for an example), whereafter the core exit pressure recovers. The phenomenon of FE itself, however, must take a finite amount of time to initiate and occur; thus, it is possible that these short time duration depressurization events may not allow enough time for FE to proceed. An initial evaluation of RELAP capability to predict this phenomenon was initiated last reporting period, and work has continued through this year.

A small transient flow excursion code (FLOWEXC), designed specifically to examine the FE phenomenon and time scales has been developed. It includes single- and two-phase frictional pressure drops, the effect of momentum changes in the channel, and correlations to predict the OSV as well as the CHF. Initial benchmarking of this code against THTL data has been performed. The code has also been used to compare to RELAP calculations, which were designed to isolate the flow excursion phenomena. These comparisons have included a relatively slow reduction in channel pressure drop (over 100 s) and a relatively slow reduction in channel exit pressure (also over 100 s). Results of these comparisons have indicated that the time scale of these events is long enough to influence the time scale of the phenomenon and that fine nodalization of the RELAP channel model is required to capture the time scale. Differences between the RELAP predictions and the FLOWEXC code predictions remain, however. This comparison is continuing in order to determine if RELAP can be effectively used in this manner and whether the FE time scales are long enough to diminish the importance of rarefaction waves as the limiting phenomena during an instantaneous pipe break event.

4.4.3 Flow Blockage Analysis

Core inlet flow blockage caused by debris within the primary coolant loop is a significant

safety issue. A local reduction of heat transfer within the fueled region of the core caused by the flow recirculation downstream of the blockage may cause some fuel melting. Efforts have focused on defining the maximum flow blockage at the core inlet that will not result in fuel damage. An initial estimate of 10 mm for this limiting blockage size (assuming an unfueled entrance length of 50 mm) was based on an extrapolation of Sparrow and Cur's (1983) experimental work in a partially blocked rectangular channel. (The experiment used mass transfer measurements at a Reynolds number of 30,000.) The extrapolation showed that the recovery to reasonable flow rates would occur at a downstream location in the channel about five times the width of the blockage. Because of the large amount of uncertainty associated with the extrapolation, an extensive effort using both analytical and experimental tools has been made to understand the flow recovery region more fully.

Both the experimental and analytical results show that the disturbance behind a blockage located on the edge of the channel persists longer than the disturbance behind a centrally located blockage. The edge blockage, therefore, represents the worst, or most conservative, case. An LDV was used to measure hydraulic parameters within a channel designed to match the ANS core cooling channels (for further description of this experimentation, see Sect. 2.4.2). The results for a 21-mm (25% of the 84-mm span) blockage placed at one side of the span show that the flow reattaches (or changes from back- to forward-flow) between 58 and 90 mm. The Fluent computational fluid dynamics analysis predicts such reattachment at 65 mm, or about three times the blockage width. The LDV results show that the flow surpasses 50% of its undisturbed velocity between 90 and 121 mm. The Fluent analysis predicts that this point will be reached at 105 mm, or about five times the blockage width. An 8.4-mm (10% of the span) blockage was also tested in an edge position. TLCs were used to measure the wall heat transfer characteristics in the region behind the blockage. These results have limited

accuracy near the side walls but show a region of reduced heat transfer extending at least 36-mm downstream from the blockage, or four times the blockage width. The Fluent results for the 10% case show reattachment occurring at 44 mm (or five times the blockage width) and the 50% velocity recovery occurring at 90 mm (or 10 times the blockage width). This 10% edge blockage will be investigated using the LDV system in the near future to provide more information about the extended length of the recovery region for this smaller blockage.

One major question must still be addressed. The hydraulic behavior within the channel has been well characterized by the work to date. The next step in the analysis will address the temperature fields within the channel (for various wall heat flux patterns) to determine whether or not the preliminary definitions of acceptable flow recovery are valid.

4.5 SEVERE ACCIDENT STUDIES

During the current reporting period, a severe accident workshop was held in November 1993 to evaluate work done to date and to help in preparing a detailed ANS severe accident program (SAP) plan to ensure timely safety certification of the ANS. Subsequent to the workshop, an initial draft of the SAP plan was completed. In addition, the severe accident analysis team pursued a fuel coolant interaction (FCI) scoping study recommended by the severe accident workshop to assess potential containment vulnerabilities from explosive FCI. This work was supplemented with assessment of core melt propagation, debris dispersion, and recriticality.

Several in-reactor experiments were successfully conducted with ANS fuel plate samples and associated modeling and analyses were conducted. The results of these FY 1994 activities are summarized in the following sections.

4.5.1 Development of the Advanced Neutron Source Severe Accident Program Plan

During FY 1994, work was performed to develop a planning document to provide a concise and coherent mechanism for presenting the ANS SAP goals, the strategy for satisfying these goals, a succinct summary of the work done to date, and an outline of future experimental and analytical work to ensure timely licensability. Guidance received from various bodies [e.g., panel members of the ANS severe accident workshop (SAW) held during November 1993, DOE orders, NRC requirements for advanced light water reactors (ALWRs) and other advanced reactors, Advisory Committee on Reactor Safeguards (ACRS) comments, and worldwide industry trends] was used to set up the ANS-relevant SAP goals and implementation strategy. Along with deterministic and probabilistic goals for containment integrity, the goals include a protection goal for experimenters and other workers routinely inside containment. The on- and off-site risk goals identified in the PDR are also included.

The strategy is to achieve closure of severe accident issues that have the potential to threaten containment during realistic bounding events, which are defined as events with an occurrence frequency greater than $\sim 10^{-6}$ /year. Based upon the current Phase 1, Level I PRA, the realistic bounding events are flow blockage of fuel element channels (including those from manufacturing defects) and RDAs. For these two initiating events, the subject of issue closure was addressed by systematically assessing the certification potential of each of eight containment failure modes or phenomena that could challenge containment integrity. The eight modes are containment bypass, explosive FCI, recriticality, containment overpressurization, molten core-concrete interactions (MCCIs), detonation of combustible gases, liner melt-through, and direct containment heating (DCH). This list represents a comprehensive compilation of the full range of potential severe-accident-related containment loads, from the slow buildup

of pressure that would accompany long-term decay heat removal inside containment without the functioning of active containment cooling features, all the way to the very rapid shock waves that would accompany explosive events from FCI. Based upon assessment of the eight areas, it was judged that FCI and recriticality were the two areas with potential for becoming certification issues and, therefore, should be focused upon for closure.

For the present discussion, recriticality may be defined as the possibility that the uranium fuel in the core debris from a hypothetical severe accident might become critical after the severe accident, perhaps after transport to another part of the coolant system. The process for issue closure would be tailored along the lines of the risk-oriented-accident-analysis-methodology (ROAAM) process (Theofanous et al. 1991) that has been applied successfully for resolving power reactor severe accident issues, such as DCH and liner melt-through. The balance of areas dealing with containment bypass, etc. were not deemed to represent certification issues for ANS. This was especially true for the liner melt-through and DCH areas, which are considered irrelevant for ANS. For the balance of the noncertification topics, work will be confirmatory in nature. Flow charts for issue (or topic) closure and for decision points, indicating whether experiments were needed for resolving key uncertainties were also developed.

The overall framework for ANS SAP work is shown in Fig. 4.5. As shown in this figure, the results of closure for individual containment failure modes and mechanisms mentioned earlier (e.g., containment bypass, FCI, recriticality, etc.) will demonstrate meeting of ANS SAP goals for two initiating events analyzed in a realistically bounding sense. This work will be performed on a separate-effects basis and will be closely tied in with methods development, experimentation, and early T/H calculations. Work performed for issue closure would represent an essential basis for making the ANS safety case to be documented in PSAR and the final safety analysis report (FSAR).

4.5.2 Assessment of Vulnerabilities from Fuel-Coolant-Interaction Loads during Flow Blockage Events

Depending upon the size and geometry of contact and the temperatures of molten materials, the interaction between molten fuel and water has the potential to generate steam via rapid vaporization of the water and also to generate hydrogen via rapid oxidation of the metallic aluminum in the molten fuel. If the interactions occur on a sufficiently short period of time and are of sufficient magnitude, shock waves can be generated, and the FCI is then characterized as explosive. The primary containment can absorb great quantities of energy before failure (on the order of gigajoules) if the loads are applied more or less uniformly throughout the pressure boundary; however, for explosive loads it becomes more difficult to demonstrate the uniform application of loads, especially if the shock waves can cause some piece of the structure to come loose and impinge upon the 16–25-mm-thick primary containment steel shell. It is difficult to quantify the effects of FCI because of phenomenological uncertainties. The greatest uncertainties include: how much of the core becomes molten; in the part of the core that becomes molten, what temperature above melting point is reached; and how much oxidation of the metallic aluminum component of the melt occurs during the FCI. The overall process for closure of FCI-related concerns is shown in Fig. 4.6.

The topic of explosive FCI received considerable discussion at the SAW, with many participants expressing the opinion that the ANS containment would not be threatened by FCI because of the very small mass of the core (thousands of times smaller than power reactor cores) and the large size of the containment (larger than most power reactor containments). Accordingly, a conclusion of the workshop was that the ANS severe accident team should perform a directed scoping study to determine if any reasonable potential exists for explosive FCI to threaten ANS containment integrity. The initial part of such a scoping study, really

Notes:

- 1) Cutoff frequencies quoted for flow blockage and depressurization events are median (not mean) values as quoted in ANS Phase I Level-1 PRA and given in Table 1.3.2.
- 2) Methods development and experimentation activities will be driven by needs identified during issue closure and PSAR/FSAR work areas. Information derived will feed back into issue closure and SAR work.

Fig. 4.5. Framework and strategy for Advanced Neutron Source severe accident safety work related to issue closure, preliminary safety analysis report, final safety analysis report, and probabilistic risk assessment.

Fig. 4.6. Advanced Neutron Source severe accident program strategy for closure of fuel-coolant-interaction issue.

the first part of the closure process of Fig. 4.6, has been completed. Although the results have not received sufficient peer review to report in detail here, they demonstrate that, if conservative bounding assumptions are made at each step of the way, it is possible to hypothesize a sufficiently energetic mechanical energy release to raise concerns about containment integrity, which leads to two conclusions: that the FCI issue cannot be dismissed out-of-hand and that more realistic analyses will be required. The more realistic analyses have been initiated but were not completed this year.

4.5.3 Severe Accident Scenario Characterization for Core Debris Dispersion and Transport

Work in this area consisted of model development and analyses for evaluating the potential for entrainment and dispersion into the reactor coolant system (RCS) during severe accident conditions. At this stage of development, it was not possible to simulate a detailed description of transient breakup while simultaneously considering thermodynamics. Rather, test-data-based information was used in a nondimensional manner (i.e., via correlation) to evaluate what particle sizes would result for a given surface destabilizing force applied to a molten particle before no more breakup can be sustained. A criterion was first set up based upon past work with droplets to evaluate entrainment potential. The theory is based upon growth of surface instabilities (Kelvin-Helmholtz type). Once entrainment occurs, the mean size of particle was evaluated using a Weber criterion corrected for fuel debris viscosity. In reality, significant variations can be expected in fluid conditions of velocity, density, and, perhaps, temperature in mixing zone(s). A probabilistic distribution that captures similar natural processes was used for this purpose.

Following this development, it was desirable to know what the distribution of debris particles would be throughout the RCS. It was recognized that the motion of individual particles in a

coolant stream would necessarily be nonuniform, always changing in direction and magnitude. Nevertheless, for ascertaining gross motion a mathematical model was developed in a one-dimensional (1-D) framework for capturing the transient rectilinear motion of particles in turbulent flow. This model accounts for inertia and virtual and viscous forces but does not account for continuous dispersion of debris particles from lift or fluidization forces.

A side calculation was conducted to evaluate the minimum velocity below which the model would break down for ANS-type conditions. The minimum drag/fluidization velocities were determined to be dependent on particle size and density, varying from as low as 0.11 m/s for aluminum particles of 0.1-mm radius to 0.68 m/s for an UAl_2 particle of 1.0-mm radius. The ANS RCS was nodalized into 60 nodes or volumes, including the heat exchangers. Fluid velocities for full flow conditions in various segments were used appropriately to account for changing diameters of pipes and volumes (such as heat exchangers). It was found that dispersed particles would be quickly entrained into the bulk stream. Assuming the whole 100-kg core mass is entrained, the distribution throughout the RCS was evaluated. Sample results are shown in Fig. 4.7. The present model will be refined to account for multidimensional effects in key components, such as heat exchangers, and is expected to provide information for evaluating debris recriticality potential.

4.5.4 Severe Accident Experimentation and Related Analysis

During FY 1994, significant progress was made in the cooperative efforts for testing of ANS fuel plates under rapid energy deposition conditions in the Japan Atomic Energy Research Institute's (JAERI's) Nuclear Safety Research Reactor (NSRR). Specifically, five miniplates were fabricated and shipped to JAERI in Tokai, Japan, and successfully subjected to power bursts in the NSRR. Three of these tests were conducted with the presence of unfueled side

Fig. 4.7. Variation of debris mass at selected locations (assumed whole core debris density = 7000 kg/m^3).

plates to represent the thermal-hydraulics of narrow rectangular channels in actual situations. Valuable data have already been obtained at different energy deposition levels to evaluate the onset of plate bowing, cracking, and melting, and the effect of geometric spacing of plates.

Mathematical models are being developed to capture the physics of these complex processes. Specifically, a 3-D model developed with the HEATING-7 code has been used to analyze tests with ANS as well as with JAERI miniplates (which have different geometry and fuel density). The model has proven very useful in identifying an important area requiring further attention (i.e.,

the rate and amount of fission energy deposition in the miniplates).

A separate work area concerned the investigation of bubble dynamics to evaluate the degree of voiding of coolant channels in those tests where side plates were used. Predictions made for the NSRR tests with side plates were in good agreement with observed data. That is, for the cases where significant vapor production was predicted, the test data indicate no surface rewetting, unlike otherwise identical tests without vapor formation wherein surface rewetting leads to a decrease in plate temperatures.

In tandem, a first-cut 2-D structural dynamics model has been developed using the CTH code. This model predicted that onset of breakup and dispersion would be closely tied to peak central temperatures high enough to begin vaporizing aluminum. Extension of this model to 3-D situations was not possible because of limitations on the workstation platform; for ANS conditions, the aspect ratios are far too large to enable reasonable simulations with CTH. For this reason, the parallel-computing version of CTH, (i.e., the parallel CTH (PCTH) code, has recently been obtained and is being implemented to work on ORNL's supercomputers.

Separately, a simple model that captures the essential physics of the situation was developed. In this model, it was proposed that fuel plate damage behavior is a function of effective dimensionless volumetric strain rate. The strain rate analogy comes from past studies on spallation during high velocity impact. Different mechanisms for strain-rate induced failure were investigated. Specifically, the model

evaluates strain rates caused by fuel heatup, expansion resulting from phase changes (i.e., melting and vaporization) and expansion of gases. The goal here is to use the concept of strain-rate-induced failure, as modeled via benchmarking against experimental data, in order to set up simple-to-use criteria that would then be utilized for predicting the onset of damage (i.e., cracking, bowing, and dispersion). Presently, this model is being applied to data taken in the transient reactor test (TREAT) facility with alloy and cermet fuel plate samples, as well as to more recent tests conducted in JAERI's NSRR. Initial results of comparisons against TREAT data for dispersion look promising.

Significant comments on ANS severe accident-related experimentation were received during the severe accident workshop and will be used, in conjunction with the issue closure work, to begin formulating the basis of experiments initially targeted towards closure of the FCI-related issues, during FY 1995.

PUBLICATIONS

FY 1994

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D. J. Alexander, "The Effects of Irradiation to $8 \times 10^{26} \text{ m}^{-2}$ on the Mechanical Properties of 6061-T651 Aluminum," *Proceedings American Society for Testing and Materials*, Sun Valley, Idaho, June 1994.

E. E. Alston, J. C. Gehin, and C. D. West, *Fuel Density, Uranium Enrichment, and Performance Studies for the Advanced Neutron Source Reactor*, ORNL/TM-12775, June 1994.

R. E. Battle et al., "Application-Specific Integrated Circuit Design for a Typical Pressurized Water Reactor Pressure Channel Trip," *Instrument Society of America 1994 Joint Power Industry Division/Electric Power Research Institute*, Orlando, Florida, June 1994.

R. E. Battle, "Safety System Upgrade Without a Computer," *TRTR Quarterly Newsletter*, February 1994.

J. J. Carbajo, "Lift Forces over a Sheared Inner Control Rod of the Advanced Neutron Source Reactor," *Proceedings American Nuclear Society 1994 Annual Meeting*, New Orleans, June 1994.

J. J. Carbajo and W. R. Hendrich, "Forces Acting During a Scram over the Inner Control Rods of the Advanced Neutron Source Reactor," *Proceedings American Nuclear Society 1994 Annual Meeting*, 252–253, New Orleans, June 1994.

N. C. J. Chen, M. W. Wendel, and G. L. Yoder, "Flow Excursion Margin Behavior During a Pipe Break in the Advanced Neutron Source Reactor," *Trans. Am. Nucl. Soc.* **70**, TANSO 70 1–458 (1994), 433–434.

N. C. J. Chen et al., *Validation and Verification of RELAP5 for Advanced Neutron Source Accident Analysis: Part I, Comparisons to ANSDM and PRSDYN Codes*, ORNL/TM-12418/V1, December 1993 (published February 1994).

N. C. J. Chen, G. L. Yoder, Jr., and M. W. Wendel, "Conceptual Design Loss-of-Coolant Accident Analysis for the Advanced Neutron Source," *Nucl. Tech. J.*, January 1994.

J. A. Crabtree, "Use of Image Processing Techniques for Heat Transfer Measurements Using Thermochromic Liquid Crystals," *Proceedings 1994 American Institute of Aeronautics and Astronautics/American Society of Mechanical Engineers Heat Transfer and Thermophysics Conference*, Colorado Springs, Colorado, June 1994.

C. C. Eberle, *Proposal for a High Flux Positron Beam Facility at the ANS*, ORNL/M-3310, February 1994.

D. K. Felde et al., *Advanced Neutron Source Thermal-Hydraulic Test Loop Facility Description*, ORNL/TM-12397, February 1994.

J. C. Gehin, J. P. Renier, and B. A. Worley, "A New Fuel Loading Design for the Advanced Neutron Source Reactor," *Proceedings American Nuclear Society Meeting on Advances in Reactor Physics*, Knoxville, Tennessee, April 1994.

M. L. Gildner, *Advanced Neutron Source Project Quality Assurance Plan, Revision 2*, ORNL/TM-11446/R2, August 1992 (published July 1994).

R. M. Harrington and C. T. Ramsey, Oak Ridge National Laboratory, and R. R. Fullwood, Brookhaven National Laboratory, "Probabilistic Risk Assessment in the Design of the Advanced Neutron Source," *Proceedings International Topical Meeting on Advanced Reactor Safety*, Pittsburgh, April 1994.

R. M. Harrington, "PRA in the Design of the Advanced Neutron Source," *DOE Risk Management Quarterly*, October 1993.

W. A. Hamilton and J. B. Hayter, Oak Ridge National Laboratory; P. D. Butler, University of Tennessee; S. M. Baker, G. S. Smith, and R. Pynn, Los Alamos National Laboratory; and L. J. Magid, University of Kentucky; "Shear Induced Hexagonal Ordering Observed in an Ionic Viscoelastic Fluid in Flow Past a Surface," *Phys. Rev. Lett.*, Vol. 72, No. 14, 2219-2222, April 1994.

W. A. Hamilton and J. B. Hayter, Oak Ridge National Laboratory, and G. S. Smith, Los Alamos National Laboratory, "Neutron Reflectometry as Optical Imaging," *J. Neut. Res.*, Vol. 2, No. 1, 1-19, 1994.

M. Ibn-Khayat, Oak Ridge Institute for Science and Education, and J. L. Anderson, R. E. Battle, and J. March-Leuba, Oak Ridge National Laboratory, *Evaluation of Secondary Coolant Control Design Alternatives and Their Effects on Heat Removal Performance*, ORNL/TM-12649, March 1994.

M. Ibn-Khayat, Oak Ridge Institute for Science and Education, and J. L. Anderson, R. E. Battle, and J. March-Leuba, Oak Ridge National Laboratory, *Evaluation of the Need for Emergency Heat Exchangers for Long Term Emergency Cooling of the Advanced Neutron Source Reactor*, ORNL/TM-12677, May 1994.

M. Ibn-Khayat, Oak Ridge Institute for Science and Education, and J. March-Leuba, Oak Ridge National Laboratory, "Plant Protection System Optimization Studies to Mitigate Consequences of Large Breaks in the Advanced Neutron Source Reactor," *Proceedings American Nuclear Society Winter Meeting*, San Francisco, November 1993.

D. H. Johnson et al., PLG, Inc.; J. Schuller and R. W. van Otterloo, N.V. KEMA; and C. T. Ramsey and M. A. Linn, Oak Ridge National Laboratory, "Examples of the Use of PRA in the Design Process and to Support Modifications," *Proceedings Probabilistic Safety Assessment and Management, Inc., Meeting*, San Diego, March 1994.

D. H. Johnson, D. C. Bley, and J. C. Lin, PLG, Inc., and C. T. Ramsey and M. A. Linn, Oak Ridge National Laboratory, "Examples of the Use of PSA in the Design Process and to Support Modifications at Two Research Reactors," *Proceedings Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operation, and Safety*, Taipei, Taiwan, April 1994.

S. H. Kim et al., "Analysis of Core-Concrete Interaction Events with Flooding for the Advanced Neutron Source Reactor," *Proceedings Fourth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations, and Safety*, Taipei, Taiwan, April 1994.

R. A. Lillie, "ANS Cold Source Neutronics Analysis," *Proceedings American Nuclear Society Meeting on Advances in Reactor Physics*, Knoxville, Tennessee, April 1994.

A. M. Ougouag, C. A. Wemple, and G. A. Rubio, Idaho National Engineering Laboratory, EG&G, Inc., "MCNP Neutronic Analysis of the FOEHN Franco-German Critical Experiment," *Proceedings American Nuclear Society Meeting on Advances in Reactor Physics*, Knoxville, Tennessee, April 1994.

A. M. Ougouag et al., Idaho National Engineering Laboratory, EG&G Idaho, Inc., *MCMP Analysis of the FOEHN Critical Experiment*, ORNL/TM-12466, October 1993 (published January 1994).

F. J. Peretz, *General Information on the Advanced Neutron Source*, ORNL/M-3264, Martin Marietta Energy Systems, Inc., Gopher.

A. B. Poole, "Simplified Design and Evaluation of Liquid Storage Tanks Relative to Earthquake Loading," *Seismic Engineering*, PVP-Vol. 275-2, American Society of Mechanical Engineers, Washington, June 1994.

A. E. Ruggles et al., *Advanced Neutron Source Reactor (ANSR) Phenomena Identification and Ranking (PIR) for Large Break Loss of Coolant Accidents (LBLOCA)*, ORNL/TM-12496, June 1994.

L. W. Scarbrough, *The Advanced Neutron Source Briefing Manual*, ORNL/M-2910/RA, January 1994.

E. A. Schneider, *A Station Blackout Simulation for the Advanced Neutron Source Reactor Using the Integrated Primary and Secondary System Model*, ORNL/M-3262, March 1994

D. L. Selby, R. M. Harrington, and P. B. Thompson, *Advanced Neutron Source (ANS) Project Progress Report, FY 1993*, ORNL-6763, January 1994.

M. Siman-Tov et al., *FY 1993 Progress Report on the ANS Thermal-Hydraulic Test Loop Operation and Results*, ORNL/M-3789, July 1994.

B. R. Smith, "Reactor Installation and Maintenance for the Advanced Neutron Source," *Proceedings Deneb Robotics User Group Meeting VII*, Troy, Michigan, October 1993.

L. A. Smith, University of Florida, and J. C. Gehin, B. A. Worley, and J. P. Renier, Martin Marietta Energy Systems, Inc., "Validation of Multigroup Neutron Cross Sections for the Advanced Neutron Source Against the FOEHN Critical Experimental Measurements," *Proceedings American Nuclear Society Topical Meeting on Advances in Reactor Physics*, Knoxville, Tennessee, April 1994.

W. F. Swinson et al., *Follow-Up Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source*, ORNL/TM-12629, November 1993 (published January 1994).

W. F. Swinson, C. R. Luttrell, and G. T. Yahr, *An Examination of the Elastic Structural Response of the Advanced Neutron Source Fuel Plates*, ORNL/TM-12712, September 1994.

R. P. Taleyarken et al., "Response of Alternate Containment Concepts Under Severe Accident Conditions for the Advanced Neutron Source Reactor," *Proceedings International Topical Meeting on Advanced Reactor Safety*, 707–723, Pittsburgh, April 1994.

T. W. Valentine, J. March-Leuba, and J. T. Mihalczo, "Monte Carlo Verification of Point Kinetics for Safety Analysis of the Advanced Neutron Source Reactor," *Trans. Am. Nucl. Soc.* **70**, TANSO 70 1–458 (1994), 366–368.

C. A. Wemple, "Detailed Heat Load Calculations for the Conceptual Design of the Advanced Neutron Source Reactor," *Proceedings American Nuclear Society Meeting on Advances in Reactor Physics*, Knoxville, Tennessee, April 1994.

C. A. Wemple, Idaho National Engineering Laboratory, EG&G Idaho, Inc., *Detailed Heat Load Calculations for the Conceptual Design of the Advanced Neutron Source Reactor*, ORNL/M-3207, December 1993 (published January 1994).

M. W. Wendel and P. T. Williams, "Two Verification Tests of RELAP5 Capabilities to Simulate Pressure-Wave Propagation for Instantaneous Pipe Breaks," *Trans. Am. Nucl. Soc.* **70**, TANSO 70 1–458 (1994), 153.

C. D. West, "The Advanced Neutron Source," *Journal of Deutsche Physikalische Gesellschaft*, November 1993.

C. D. West, "Advanced Neutron Source—eine neue Anlage für die Neutronenforschung," *Phys. Bl.* **50**, Heft 1, p. 10, January 1994.

C. D. West, "Studies of the Impact of Fuel Enrichment on the Performance of the Advanced Neutron Source Reactor," *Proceedings 1994 Reduced Enrichment for Research and Test Reactors International Meeting*, Williamsburg, Virginia, September 1994.

C. D. West, *IGORR News*, Issue No. 5, ORNL/M-3293, February 1994.

G. L. Yoder et al., *Steady-State Thermal-Hydraulic Design Analysis of the Advanced Neutron Source Reactor*, ORNL/TM-12398, May 1994.

G. L. Yoder, "Conceptual Design Loss-of-Coolant Analyses for the Advanced Neutron Source," *Nucl. Tech. J.*, January 1994.

REFERENCES

- Alston, E. E., Gehin, J. C., and West, C. D. June 1994. *Fuel Density, Uranium Enrichment, and Performance Studies for the Advanced Neutron Source Reactor*, ORNL/TM-12775, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Bari, R. A., et al. December 31, 1993 (draft). "Advanced Neutron Source Enrichment Study," to be published at Brookhaven National Laboratory, Upton, New York.
- Beck, R. G., Brown, E. S., and Rainwater, J. H. September 1958. *ETR Core Hydraulics*, IDO-16464, Phillips Petroleum Company, Idaho Operations Office, U.S. Atomic Energy Commission, Washington.
- Beck, R. G. November 7, 1958. *Hydraulics of Modified Fuel Elements for ETR*, IDO-16465, Phillips Petroleum Company, Idaho Operations Office, U.S. Atomic Energy Commission, Washington.
- Bergles, A. E., and Rohsenow, W. M. 1962. *Forced-Convection Surface Boiling Heat Transfer and Burnout in Tubes of Small Diameter*, DSR Report No. 8767-21, Dept. of Mechanical Engineering, Massachusetts Institute of Technology, Cambridge, Massachusetts.
- Booth, R. S. August 1994. *Advanced Neutron Source Plant Design Requirements*, ORNL/TM-11625/R5, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Brake, T. E. June 10-14, 1990. "Network Communications at the Diablo Canyon Power Plant Integrating Plant Process Data," *Trans. Am. Nucl. Soc. Annual Meeting*, Vol. 61, 311, American Nuclear Society, Washington.
- Breismeister, J. F. September 1986. *MCNP Version 3b, Monte Carlo Neutron and Photon Transport Code System*, LA-7396-M, Rev. 2, Los Alamos National Laboratory, Los Alamos, New Mexico.
- Computational Modeling Power for Scientists and Engineers 1991. *FLOW-3D User's Manual*, FSI-91-00-1, Flow Science, Inc., Los Alamos, New Mexico.
- Copeland, G. L. et al. August 1989. *Advanced Neutron Source Final Preconceptual Reference Core Design*, ORNL/TM-11234.
- Copley, J. R. D. 1990. "An Analytical Method to Characterize the Performance of Multiple Section Straight-Sided Neutron Guide System," *Nucl. Instr. and Meth.* A287, 363.

- Copley, J. R. D. 1991. "Acceptance Diagram Analysis of the Performance of Vertically Curved Neutron Monochromators," *Nucl. Instr. and Meth.* **A301**, 191.
- Costa, J. 1967. *Measurement of the Momentum Pressure Drop and Study of the Appearance of Vapor and Change in the Void Fraction in Subcooled Boiling at Low Pressure*, ORNL/TR-90/21, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Crow, M. L., and Steyerl, A. 1994. "Investigation of Ultra Cold Neutron Production Methods for the Advanced Neutron Source," to be published *Nucl. Inst. and Methods*.
- Currat, R. 1990. "The Efficiency of Vertically Bent Neutron Monochromators," *Nucl. Instr. and Meth.* **170**, 21.
- Engineering Mechanics Research Corporation. September 1992. NISA II, Numerically Integrated Elements for System Analysis, Troy, Michigan.
- EPRI. May 1987. *Guidelines for Specifying Integrated Computer-Aided Engineering Applications for Electric Power Plants*, EPRI Report NP-5159S, Electric Power Research Institute, Palo Alto, California.
- Felde, D. K., et al. February 1994. *Advanced Neutron Source Reactor Thermal-Hydraulic Loop Facility Description*, ORNL/TM-12397, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Fluent, June 1993. *Fluent User's Guide*, Version 4.2, Flow Science, Inc., Lebanon, New Hampshire.
- Ibn-Khayat, M., et al. 1994. *Evaluation of the Need for Emergency Heat Exchangers*, ORNL/TM-12677, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- ILL (Institut Laue-Langevin) 1994. *Annual Report 1994*, Institut Max Von Laue-Paul Langevin, Grenoble, France.
- Kohn, W., Chair. January 1993. *Neutron Sources for America's Future, Report of the Basic Energy Sciences Advisory Committee Panel on Neutron Sources*, DOE/ER-0576P, Office of Energy Research, U.S. Department of Energy, Washington.
- March-Leuba, J. and Ibn-Khayat, M. June 1990. "Dynamic Modeling of the Advanced Neutron Source Reactor," *Trans. Am. Nucl. Soc.* **61**, 386-388.
- Merryman, L., Ballon, E., and Christie, B. March 1994. "Predictability vs Availability at the High Flux Isotope Reactor," *Proceedings International Conference Devoted to the Advancement of Systems and Processes*, San Diego, California.
- Meyer, M. A., and Booker, J. M. January 1990. *Eliciting and Analyzing Expert Judgement*, NUREG/CR-5424 (LA-11667), University of California at Santa Barbara, Santa Barbara, California.

- Mirzadeh, S., et al. November 1992. *Projected Medical Radioisotope Production Capabilities of the Advanced Neutron Source (ANS)*, ORNL/TM-12010-S, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Mosleh, A., Bier, V., and Apostolakis, G. August 1987. *Methods for the Elicitation and Use of Expert Opinion in Risk Assessment*, NUREG/CR-4962 (PLG-0533), PLG, Inc., Newport Beach, California.
- Nogita, K., and Une, K. 1994. "Radiation-Induced Microstructural Change in High Burnup UO_2 Fuel Pellets," *Nucl. Instr. and Met. in Phys. Res., B*, **91**, 301–306.
- NRC (Nuclear Regulatory Commission) 1984. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, NUREG-0800, U.S. Nuclear Regulatory Commission, Washington.
- Pawel, R. E., et al. April 1993. *The Corrosion Behavior of 8001 Al Under Heat-Transfer Conditions in an Aqueous Loop System*, ORNL/NPR-92/65, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Pawel, R. E., et al. June 1990. *The Development of a Preliminary Correlation of Data on Oxide Growth on 6061 Aluminum Under ANS Thermal-Hydraulic Conditions*, ORNL/TM-11517, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Pawel, R. E., et al. 1990. "Fuel Cladding Corrosion Studies for the Advanced Neutron Source," *Trans. Am. Nucl. Soc.* **61**, 388.
- Pawel, R. E., et al. 1991. "The Corrosion of 6061 Aluminum Under Heat Transfer Conditions in the ANS Corrosion Test Loop," *Oxid. of Metals*, **36** (1/2), 175.
- Pawel, R. E., et al. March 1993. "The Corrosion Behavior of 8001 Al Under Heat Transfer Conditions in an Aqueous Loop System," *NACE CORROSION '93*, Paper No. 188, New Orleans, Louisiana.
- Pawel, R. E., Felde, D. K., and McFee, M. T. 1993. "The Development of an Improved Correlation for Corrosion Product Growth on Aluminum Alloy Fuel Cladding for the Advanced Neutron Source," p. 467 in *6th Intl. Symp. on Environmental Degradation of Materials in Nuclear Power Systems*, R. E. Gold and E. P. Simonen, eds., TMS/NACE/ANS, San Diego.
- Petukhov, B. S. 1970. "Heat Transfer and Friction in Turbulent Pipe Flow with Variable Physical Properties," *Adv. in Heat Trans.* **6**, 528.
- Poole, A. B., Battiste, R. L., and Clinard, J. A. January 1993. *Final Report Pipe Break Testing of Primary Loop Piping Similar to Department of Energy's New Production Reactor-Heavy Water Reactor*, ORNL/NPR-92/64, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Rest, J., and Hofman, G. L. 1991. "Kinetics of Recrystallization and Fission-Gas-Induced Swelling in High Burnup UO_2 and U_3Si_2 Nuclear Fuels," pp. 443–456 in *Fundamental Aspects of Inert Gases in Solids*, S. E. Donnelly and J. H. Evans, eds., Plenum Press, New York.

- Rest, J., and Hofman, G. L. 1994. "Dynamics of Irradiation-Induced Grain Subdivision and Swelling in U_3Si_2 and UO_2 Fuels," *J. Nucl. Mater.* **210**, 187–202.
- Rhoades, W. A., and Childs, R. L. May 1988. "The DORT Two-Dimensional Discrete Ordinates Transport Code," *Nucl. Sci. Eng.* **99**(1), 88–89.
- Saha, P., and Zuber, N. 1974. "Point of Net Vapor Generation and Vapor Void Fraction in Subcooling Boiling," *Proc. of 5th Intl. Heat Trans. Conf.*, Tokyo **IV**, 175–179.
- Selby, D. L., Harrington, R. M., and Peretz, F. J. April 1990. *Advanced Neutron Source (ANS) Project Progress Report*, ORNL-6574, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Selby, D. L., Harrington, R. M., and Peretz, F. J. February 1991. *Advanced Neutron Source (ANS) Project Progress Report*, ORNL-6656, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Selby, D. L., Harrington, R. M., and Peretz, F. J. January 1992. *Advanced Neutron Source (ANS) Project Progress Report—FY 1991*, ORNL-6696, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Selby, D. L., Harrington, R. M., and Peretz, F. J. January 1993. *Advanced Neutron Source (ANS) Project Progress Report—FY 1992*, ORNL-6723, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Selby, D. L., Harrington, R. M., and Peretz, F. J. January 1994. *Advanced Neutron Source (ANS) Project Progress Report—FY 1993*, ORNL-6763, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Siman-Tov, M., et al. December 1991. "Thermal-Hydraulic Correlations for the Advanced Neutron Source Reactor Fuel Element Design and Analysis," *Proc. of 1991 ASME Winter Annual Meeting*, American Society of Mechanical Engineers, Washington.
- Siman-Tov, M., et al. August 1993. "Experimental Investigation of Thermal Limits in Parallel-Plate Configuration for the Advanced Neutron Source Reactor," AICHE Symposium Series Vol. 89, *Proc. of 29th Natl. Heat Trans. Conf.*, Atlanta.
- Siman-Tov, M., et al. 1994. "Experimental Study of Static-Flow Instability in Subcooled Flow Boiling in Parallel Channels," to be published in *Proc. of 4th ASME/JSME Thermal Eng. Joint Conf.*, Maui, Hawaii.
- Sparrow, E. M., and Cur, N. August 1983. "Maldistributed Inlet Flow Effects on Turbulent Heat Transfer and Pressure Drop in a Flat Rectangular Duct," *ASME J. of Heat Trans.*, **105**, 527–535.
- Stromquist, W. K., and Sisman, O. June 1948. *High Flux Reactor Assemblies Vibration and Water Flow*, ORNL-50, Union Carbide Corp., Oak Ridge National Laboratory.

- Swinson, W. F., et al. November 1993. *Followup Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source*, ORNL/TM-12629, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Theofanous, T. G., et al. August 31, 1991. *The Probability of Liner Failure in a Mark-I Containment*, NUREG/CR-5423, U.S. Nuclear Regulatory Commission/UCSB, University of California at Santa Barbara, Santa Barbara, California.
- U.S. Department of Energy, Type B Investigation of the May 31, 1994 Fire and Contamination at the TRISTAN Experiment, High Flux Beam Reactor, Brookhaven National Laboratory, Upton, New York.
- West, C. D. August 1989. *Optimization of the PS-2 Core Configuration for the ANS Reactor*, ORNL/M-898, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Whittle, R. H., and Forgan, R. 1967. "A Correlation for the Minima in the Pressure Drop vs Flow-Rate Curves for Sub-Cooled Water Flowing in Narrow Heated Channels," *Nucl. Eng. Design* **6**, 89-99.
- Yoder, G. L., et al. 1992. "The Effect of Aluminum Corrosion on the Advanced Neutron Source Fuel Design," *Nucl. Eng. Design* **136** 401.
- Vondy, D. R., Fowler, T. B., and Cunningham, G. W. 1990. III, *The Bold Venture Computation System for Nuclear Reactor Core Analysis*, Version III, ORNL-5711, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.
- Warren, J. E. March 1994. Gilbert/Commonwealth, Inc., personal communication to J. E. Cleaves, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory.

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