SARM

SAFETY ANALYSIS AND RISK MANAGEMENT

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ACTION PLAN FOR UPDATED CHAPTER 15 ACCIDENT ANALYSIS IN THE SRS PRODUCTION REACTOR SAR (U)

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1.0 Introduction

An upgrade of the Chapter 15 Accident Analysis in the SRS Production Reactor Safety Analysis Report (SAR) in Reference 1 is required prior to K-Restart (Reference 2). This Action Plan covers the upgrade to SAR Chapter 15 and delivery by WSRC to DOE by June 1990. In addition, it constitutes the Action Plan for the non-LOCA transient analyses that will be performed by the SRL/SARM/Reactor Safety Research Section (RSRS) for Chapter 15. The Action Plan for the LOCA/non-LOCA transient analyses that will be performed by the SRL/NRTSC/Nuclear Engineering Section (NES) is addressed in Reference 3. The proposed scope of transients to be assessed by RSRS and NES in the upgraded Chapter 15 is in Table 1.

2.0 Summary

This report describes the Action Plan for the upgrade of the Chapter 15 Accident Analysis in the SRS Production Reactor SAR required for K-Restart. This Action Plan will be updated periodically to reflect task accomplishments and issue resolutions.

3.0 Action Plan

The Action Plan for upgrading Chapter 15 Accident Analysis in the SRS Production Reactor SAR consists of the two key elements: objectives and maintenance.

3.1 Objectives

Pre-Restart Program objectives include:

1. Identifying the format of the upgraded Chapter 15 Accident Analysis for the SRS production reactor SAR.
2. Identifying the design basis events (DBEs) representing extreme but credible safety challenges to the various mitigation systems that will be assessed in the upgraded Chapter 15 of the SAR, the assessment criteria and assigning responsibility between the Nuclear Engineering and Reactor Safety Research Sections of SRL.
3. Providing a schedule for assessment of the DBEs in Chapter 15.
4. Providing a schedule for radiological dose assessment of the DBEs in Chapter 15 and completion of these assessment activities.
5. Preparation and review of the various Chapter 15 amendments.
6. Identifying and justifying appropriate input data required for assessment of the

non-LOCA DBEs in Chapter 15 by RSRS.

7. **Development of "core safety limits" and the associated uncertainty treatment.**
8. **Specification of the functional requirements for the non-LOCA safety analysis codes.**
9. **Certification of the methodology for appropriate RSRS/NES transient analyses.**
10. **Identifying and analyzing supporting calculations required for Technical Specifications, safety limits, and setpoints. Provide a focal point for addressing uncertainties in the setpoints.**
11. **Identifying the format of QA documentation to be used to assess each DBE including a description of the postulated event, the protection provided, the assessment criteria, the reported SAR results, and details of the analysis (calculational procedure, information from other groups [i.e., plant, nuclear, and thermal-hydraulic], assumptions used, description of code input, and a checklist of critical parameters).**
12. **Completion of the appropriate QA documentation.**
13. **Review of the DBEs to be assessed in the upgraded Chapter 15 of the SAR , the upgraded format for Chapter 15, and the schedule for completion of the upgraded Chapter 15 amendments with the DOE, the Reactor Restart Division, and the ESH & QA Division.**

Developing a structure and format for transmittal of the limits documents and developing a charge design safety analysis checklist (CDSAC) for generic core charge calculations (WISR-43) are two K-restart activities which are closely associated with preparation of the SAR. However, these activities not part of the scope for this Action Plan.

3.2 Maintenance

Task descriptions and schedules in this Action Plan will be periodically updated to reflect completion of the tasks.

4.0 Tasks

This section describes the various tasks necessary for the upgrading the Chapter 15 Accident Analysis in the SRS production reactor SAR.

4.1 Upgraded Chapter 15 Accident Analysis Format

The format of the upgraded Chapter 15 Accident Analysis will be based on REG GUIDE 1.70 (Reference 4) and will address NUREG-0800 (Reference 5) requirements for commercial light water reactors (LWRs) as appropriate to SRS production reactors. Key elements of this task are:

1. Reviewing the NUREG-0800 requirements for each DBE analyzed.
2. Selecting the appropriate requirements for Chapter 15 of the SRS production reactor SAR.

4.2 Design Basis Events for Upgraded Chapter 15

A proposed scope of Design Basis Events (DBEs) to be included in the upgraded Chapter 15 is in Table 1 along with the responsible SRL organization. Key elements of this task are:

1. Reviewing the selection of the DBEs in Table 1 for completeness and adequacy, considering the Charlotte and WISR reviews, and any outstanding issues, and assigning responsibility for any new ones added. DBEs will specifically include performance of Supplementary Safety System during loss of AC power and loss of fuel reactivity insertion.
2. Identifying DBEs requiring operator actions. Assessment of these DBEs will be scheduled first to confirm that the actions are prescribed by procedures. If not, the actions will be incorporated into procedure revisions and training provided to ROD personnel prior to K-restart. The operator actions will be reviewed to ensure that they are credible, in the time frame available and the expected physical and psychological environment.
3. Defining each DBE including the sequence of events, appropriate initial conditions, protective systems available, and the required operator actions accounted for in the DBE assessments. The DBEs will be defined to include anticipated transient events with failure of safety rods to scram (i. e., those for which Confinement Protection Limits [CPLs] are applicable).
4. Establishing how DBEs should be categorized (e.g., with respect to likelihood of occurrence as defined by General Design Criteria for LWRs [Appendix A to 10CFR50], or as defined by ANSI N18.2, or some equally valid scheme.
5. Categorizing each DBE as established in 4.2.4 above.

4.3 DBE Assessment

This subsection specifies the criteria that will be used to assess the DBEs and discusses the assessment strategy to be used for the DBE analyses.

4.3.1 Assessment Criteria

Assessment (design) criteria for the upgraded Chapter 15 include flow instability, burnout risk, overpressure (CPLs), Bingham pump and elbow cavitation, and bulk moderator temperature. This task will identify the appropriate criteria for assessing each DBE.

4.3.2 Assessment Strategy

An assessment of each DBE in the upgraded Chapter 15 will be provided either quantitatively with approved¹ transient analysis computer codes or hand calculations or qualitatively by a safety evaluation demonstrating adequate safety margin. Except for LOCA, transient analyses will be performed at a power level greater than or equal to that permitted by ECS power limits which will ensure adequate safety margin. Key elements for this task are:

1. Identifying the method of assessment (computer analysis, hand calculation, or safety evaluation) for each DBE in the upgraded Chapter 15.
2. Assessing the DBEs in Table 1 and developing upgraded Chapter 15 amendments.

4.3.3 Dose Assessments

A radiological dose analysis will be provided by RSRS for DBEs described in the upgraded Chapter 15 of the SAR. If there are no radiological consequences associated with a DBE, the assessment will simply contain a statement indicating that containment of the activity was maintained within the fuel cladding.

4.4 Input for Chapter 15 Analysis

Reactor operations, physics, and thermal-hydraulic data are required for the non-LOCA analyses in the upgraded SAR Chapter 15. Key elements of this task are:

1. Specifying the Reactor operations, physics, and thermal-hydraulic data required
 1. As used here, "approved codes" mean those meeting WSRC QA requirements for critical application codes, except that codes with an established record of use at SRS for design and safety analyses need not be completely certified (See Reference 8). They must, however, have a certification plan in progress, and there must be written justification that there is reasonable basis to proceed.

for the non-LOCA analyses in the upgrade SAR Chapter 15. A preliminary specification for the non-LOCA analysis input is shown in Table 2.

2. **Developing** conservative preliminary input data for the upgraded Chapter 15 analyses. The preliminary data will serve two key functions: 1) provide conservative analyses to adapt analytical tools, train analysts, identify critical areas for detailed effort, and develop amendments for the upgraded Chapter 15, and 2) provide a vehicle for discussion to properly define the final input data package.
3. Providing an assessment of the impact of the Technical Baseline Program on the preliminary input data and recommendations for timely inclusion of pertinent data so that the safety analyses reflect plant configuration for K-Restart.
4. Developing approved input data for the calc-notes associated with the upgraded Chapter 15 analyses that has met applicable QA requirements.
5. Defining the flow paths and interface agreements for the input data.
6. Defining the schedule(s) for receiving the preliminary and approved data.

4.5 Core Safety Limits and Uncertainty Treatment

Flow instability (FI) limits are the most important "core safety limits" for the SRS production reactors (Reference 3). They apply to non-LOCA transients and are applied as conservative interpretations of the LOCA criterion to maintain coolable geometry. In addition, static calculations of these limits can be applied directly to the non-LOCA transients (see Appendix I). Data derived using the methodology developed for the LOCA-FI program will be used to account for non-axisymmetric effects and uncertainty. Initially, LOCA-FI data will be used directly with a substantial margin (30%) applied to their impact to provide a conservative estimate. These data may be updated before K-restart. The "core safety limits" are assembly inlet temperature versus assembly power for which there is a specified probability/confidence level that no assembly in the reactor core undergoes FI. Key elements of the task are:

1. Establishing the appropriate probability/uncertainty level for the "core safety limits"; i. e., is 95% probability at 95% confidence level the appropriate choice?
2. **Developing** the appropriate uncertainty factors to be included in the "core safety limits"; e. g., should eccentricity and hardware loss coefficients be included and heated wall effects excluded?
3. Generating the "core safety limits".

4.6 Analysis Methodology

This section discusses the computer codes to be used for the LOCA and non-LOCA analysis of DBEs in the upgraded Chapter 15 amendments.

4.6.1 LOCA Analysis

Safety analysis of LOCA transients will be performed in SRL/NRTSC/NES. The analyses will primarily use the RELAP, TRAC, LLAP(AA3) and FLOWTRAN codes. Application of these and supporting codes to LOCA transients is discussed in Reference 3.

4.6.2 Non-LOCA Analysis

Safety analysis of non-LOCA transients will be performed primarily in SRL/SARM/RSRS. Non-LOCA transients involving rapid coolant flow perturbations require use of the set of codes discussed in section 4.6.1 and will be performed by SRL/NRTSC/NES.

4.6.2.1 Non-LOCA Analysis Codes

Safety analysis methodology for non-LOCA transients is being assessed. Currently, two non-certified codes exist for non-LOCA analysis applications: the AA3 code and the MARY code. Both codes model neutronic and engineering aspects of the SRS reactors, including the reactor core, the primary coolant loop, the shutdown systems, and essential parts of the protective instrumentation system. AA3 has been previously used for calculation of the Confinement Protection Limits for the Savannah River Reactors and for both routine and specialized transient analysis pertaining to the SRS production and test reactors. MARY was used for safety analysis for the Westinghouse Independent Safety Review Committee and for independent analysis of reactivity transients. It is now being certified (see qualification schedule attached as Appendix II). The decision on whether to use AA3 or MARY will be made early in 1990, consistent with the needs for evaluation of the SAR non-LOCA events. Also, software functional requirements will be generated as an aide to making that choice, and to determine if any further development is appropriate. Note that the MARY code is scheduled to be certified and benchmarked prior to startup. AA3 has a prior history of application to calculation of power limits for the Savannah River Reactors, and therefore need not be completely certified prior to use. However, it must have a certification program underway, and a documented justification of the basis for proceeding with it.

4.6.2.2 Non-LOCA Analysis Approach

The strategy to be used for the non-LOCA analysis is independent of the code to be used, or even code versus hand calculations.

Each DBE will be shown to meet criteria applicable to that event at a power level higher than permitted by the ECS power limits. That demonstration may be computer or hand calculation, or qualitative by showing that some other analyzed event with the same criteria and protective actions is necessarily more limiting.

All anticipated operational occurrences must be shown to meet the flow instability limit (see Appendix I on specification for these limits). In addition, all postulated accidents will be shown to meet the FI limit, even though fuel damage and radiological dose to personnel, rather than FI, are the actual criteria of concern. (FI is a conservative definition of the latter limits.)

Further, all anticipated operational events will be assessed for failure of safety rods to scram; i.e., with shutdown from automatic actuation of the Supplemental Safety System. The applicable criteria are limited fuel damage, preservation of confinement system integrity, limited radiological release, preservation of long term core cooling, and no propagation to a more severe accident. (For K-restart, the limited fuel damage will be zero). These criteria are embodied in the Confinement Protection Limits, previously taken as the steaming rate that does not exceed yield stress in the roll anchors on the six inlet pipes to the plenum. For the restart analyses, these events will be shown to cause no bulk steaming at all, even considering FI, and therefore are well within the allowable limits. This approach does not attempt to define and defend the actual limit for these events, but merely shows the criteria are met.

4.7 Technical Specification Support Calculations

Technical specifications being developed under the Technical Specification Improvement Program define the envelope or upper bound of authorized operation on the SRS production reactors. They specify the conditions, safe boundaries, and procedures under which all activities are to be conducted. Safety limits are established for primary reactor operating variables so that steady-state operation within these limits will not cause damage to the reactor core. Limiting conditions for operation are established to restrict operating variables that could have a significant effect on reactor safety under transient conditions. Technical support is required to ensure that the Technical Specifications are consistent with assumptions in the derivation of Power Limits and the associated safety analyses. Key elements of this task are:

1. **Defining** the technical support tasks and responsible organizations necessary to **ensure** technical specifications are consistent with assumptions in the derivation of reactor power limits and the associated safety analyses.
2. Performing the assigned tasks.

4.8 QA Documentation for DBE Analyses

Calculation-notes (Reference 6) will be used for non-LOCA analyses and other equivalent SRL-approved procedures (Reference 7) will be used for LOCA analyses to provide QA documentation for each of the DBE analyses.

4.9 Chapter 15 Preparation and Review

Preparation of the various upgraded Chapter 15 amendments according to the SRS style guide in Reference 8 is required once the safety analyses have been completed and drafts of the upgraded Chapter 15 amendments developed. In addition, review, approval and issuance is required. Key elements for this task are:

1. Preparing the draft version of the upgraded Chapter 15 amendments according to the SAR style guide.
2. Reviewing the upgraded Chapter 15 amendments with appropriate WSRC Divisions and the DOE and obtaining appropriate approvals per Reference 9 and 10.
3. Issuance of the upgraded Chapter 15 amendments.

5.0 Post-Restart Objectives

The Post-Restart objectives include:

1. Long-term improvements for upgrading DBE analyses. One item is the development of standardized procedures (safety analysis standards) for performing each DBE that are similar to those used for commercial LWRs. Attributes of safety analysis standards are shown in Table 3.
2. Long-term improvements to the current thermal-hydraulic limits system.
3. Design Reviews for each DBE, as appropriate, to justify removing unnecessary safety margin from the assessments while retaining overall conservatism and defensibility.
4. Increasing reactor power levels as much as possible while still ensuring safety of operation and protection of both personnel and the environment.

6.0 Schedules

Figure 1 shows the schedule for obtaining a defensible upgrade for Chapter 15 of the SRS production reactor SAR for K restart. The schedule is based on a normal 5 day/40

hour week from Monday through Friday.

Boxes with rounded corners represent milestone items in the overall task schedule.

Boxes with square corners represent scheduled tasks.

Similar boxes with internally rounded corners represent detailed tasks (i.e., tasks composed of many subtasks). Details of these tasks are not included in this report; however, they will be used by management to meet over schedule requirements.

Dates in the upper left hand corner represent the earliest start date; dates in the upper right hand corner represent the earliest finish date. Underlined dates represent fixed dates in the schedule (see milestone box above).

The resource(s) associated with each task are shown above each task. These

resources are subject to change as personnel are reassigned to meet various restart commitments.

The duration of the task in work days is shown in the lower left hand corner. The elapsed calendar days for the specified task are shown in the lower right hand corner. For detailed tasks the duration is set equal to the number of elapsed calendar days. Bold lines in Figure 1 represent the critical path tasks for the Action Plan.

7.0 References

1. Savannah River Site Production Reactor Safety Analysis Report, WSRC-SAR-RX-K, WSRC-RP-89-383-VOL IX, July 17, 1989.
2. Letter, R. E. Tiller to J. S. Moore, "Comments on Interim Safety Analysis Report", dated August 7, 1989.
3. Patrick, L. W., Action Plan to Upgrade Thermal-Hydraulic Limits for Savannah River Site Production Reactors (U), RC-RP-89-373, June 30, 1989.
4. Reg. Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 3, November 1978.
5. Standard Review Plan, NUREG-0800.
6. SA&RM Calc-Note Procedure. Revision 1, SRL-RSC-890034, September 18, 1989.
7. Zagrodnik, A. A., QA Procedures for the Nuclear Technology and Scientific Computations Program Management Team, DPSTM-88-700-9, August 1988.
8. SAR Style Guide for the DuPont SRP SAR Update Project, Bechtel National, Inc. and Westinghouse Electric Corporation, December 1987, Revision A.
9. WSRC-T1, 2.02, Production Reactor Safety Analysis Report Change Request Procedure, 10/23/89.
10. ~~WSRC-T1, 2.03, Production Reactor Safety Analysis Report Revision Procedure, 10/23/89.~~

Figure 1. Master Action Plan Schedule for Upgraded Chapter 15 Accident Analysis

Table 1. Proposed Design Basis Events for Upgraded Chapter 15

<u>Transient Categories and DBEs</u>		<u>Responsible SRL Organization</u>
I.	Increase/Decrease In CW Heat Removal	
I.1	Increase/Decrease in CW temperature	RSRS
I.2	Increase/Decrease in CW flow	RSRS
I.3	Loss of CW Pump Power (All 190's)	RSRS
I.4	Loss of CW Supply to Reservoir (186 Basin)	RSRS
I.5	Loss of CW Inventory with PW Pumping	RSRS
I.6	Loss of CW Inventory without PW Pumping (classical LOPA)	NES
II.	Decrease In PW System Flow Rate	
II.1	Loss of PW Pump Power (AC motors)	RSRS
II.2	Loss of CW/PW Pump Power (blackout)	RSRS
II.3	Rotovalue Closure	NES
II.4	Pump Shaft Break (rotating/locked impeller)	NES
II.5	Loss of Blanket Gas Pressure (Slow/Sudden)	RSRS
II.6	Loss of Control Rod Cooling	RSRS/NES
II.7	Partial Fuel/Target Blockage	RSRS
III.	Reactivity and Power Distribution Anomalies	
III.1	Single Control Rod Withdrawal	RSRS
III.2	Single Control Rod Insertion	RSRS
III.3	Rapid Partial Length Rod Insertion	RSRS
III.4	Gang Control Rod Withdrawal	RSRS
III.5	Loss of Fuel	RSRS
III.6	Loss of Target	RSRS
III.7	Reloading Error	RSRS
III.8	Startup Accident	RSRS
III.9	Light Water Addition	RSRS
IV.	Increase In PW System Inventory	
IV.1	Inadvertent actuation of ECS	RSRS
IV.2	Inadvertent actuation of SSS	RSRS
V.	Decrease In PW System Inventory	
V.1	Loss of Coolant Accident (DEGB and large)	NES
V.2	Small line breaks (& spectrum)	NES
V.3	Heat exchanger tube leak	RSRS

Table 2. Typical Input Data Required for Non-LOCA Analyses

	Responsible Organization
I. Reactor Operations Data	RRD/ROD/RED/RT
<ul style="list-style-type: none"> - Operating temperature vs power & river water temperature (max. and min.) - Max. and Min. flow rates into the reactor plenum - CWS and PWS steady-state flows - CWS and PWS flow coastdown rates for loss of AC power - Max. and Min. pressures throughout the PWS - CWS and PWS geometric data (volumes and lengths) - Reactor tank mixing patterns - Scram instrumentation accuracy and time delays (max. and min.) 	
II. Physics Data	SRL/NRTSC/SCS/RPG
<ul style="list-style-type: none"> - Reactivity balance for minimum shutdown margin, BOC & EOC - Reactivity coefficients <ul style="list-style-type: none"> o BOC most positive & most negative o EOC most negative o Fuel o Target o Coolant (active and inactive) - Safety rod worth vs time - SSS reactivity worth vs time (until ink into moderator) - Decay heat vs time - Delayed neutron fractions & decay constants, prompt neutron lifetime, & neutron sources - Radial power & flow histogram - Max. incremental rod worth (single and gang) - Abnormal radial power distributions when beyond design shapes shapes for single rod withdrawal and part-length rod drop - Max. reactivity worth & peaking factor for part-length rod drop - Power deposition in various segments (fuel, target, coolant and moderator) 	
III. Thermal-Hydraulic Data	SRL/NRTSC/NES
<ul style="list-style-type: none"> - Core steady-state safety limits for flow instability (& correlations) - Design axial power shape - Design radial power & flow shape (histogram) - Nominal sub-assembly channel flow fractions - Design thermal conductivity properties for fuel and target - D2O fluid properties - Assembly geometric data 	

Table 3. Proposed Structure of Safety Analysis Standards for DBEs

1. DBE Description
2. Description of Protection
 - 2.1 Scram Systems
 - 2.3 Engineered Safety Features
 - 2.3 Operator Actions
3. Assessment Criteria
4. Reported Results
 - 4.1 Listing of plant parameters versus time in SAR Analysis
5. Analysis
 - 5.1 Computational Procedure
 - 5.2 Information from Other Groups
 - 5.3 Assumptions Used in Analysis
 - 5.3.1 Initial Conditions
 - 5.3.2 Simulation of DBE
 - 5.3.3 Protection System Features Assumed
 - 5.3.4 Physics Parameters
 - 5.4 Description of Code Input
 - 5.5 Checklist of Critical Parameters

APPENDIX I - Core Safety Limits

INTRODUCTION: The term "Core Safety Limit", as used below, can also be termed the 'fuel design limit'; or the "specified acceptable fuel design limit" in the parlance of the General Design Criteria (Appendix A to 10CFR50).

By whatever term, these limits are those selected by the designer and approved by the regulator as conservative limits which, if not exceeded, will provide a high degree of assurance that no fuel damage occurs. That is, fuel integrity (approach to melting) may be the ultimate concern, and it is conservatively assured by fuel design limits that permit no significant increase in fuel clad temperature.

The flow instability limits are the most important core safety limits for the SRS reactors. They apply strictly only to anticipated transients. However they are also applied as conservative interpretations of the LOCA criterion (to maintain coolable geometry), for some other postulated accidents.

CORE SAFETY LIMITS: Assembly inlet temperature vs assembly power (tables or plots) for the locus of steady-state points for which there's a 95% probability at the 95% confidence level that no assembly in the core undergoes flow instability. (The 95/95 specification needs to be reviewed. It was suggested here because of the close parallel with PWR core safety limits based on Departure from Nucleate Boiling.)

These calculations should be based on:

- o Nominal design assembly flow and gas blanket pressure,
- o Beginning and end of charge life (BOC & EOC),
- o Design axial power shape and radial power histogram, and

Documentation should include:

- o Nominal assembly flow,
- o Nominal blanket gas pressure, and
- o ~~Pressure~~ in the assembly at the point of closest approach to FI (i.e., minimum Stanton number).

The documentation should confirm that:

- o effects of variation in pressure follow the saturation line (i.e., that maintaining constant subcooling at constant power and flow maintains a

constant safety margin to FI). If this cannot be accomplished, the variation of core safety limits with pressure should be defined.

- o ~~core~~ safety limits are dependent only upon the power to flow ratio, as modified by the pressure change in the channel, at the point of minimum Stanton number, caused by flow changes. The local pressure vs flow (at the location of the minimum Stanton number) shall also be provided.

Uncertainty treatment shall be as follows:

INCLUDE all uncertainties relevant to steady-state evaluation. These include the following:

- o Model Variables applicable in steady-state: Hardware loss coefficients, tube power fractions, axial heated tube length, and D2O properties;
- o Model Omissions: Eccentricity, fuel concentration, azimuthal tilt, rib effect, lithium non-uniformities;

EXCLUDE all uncertainties based on transient effects. These include the following:

- o Model uncertainties: Heat transfer coefficient, tank bottom pressure transient, and plenum pressure transient;
- o PWS pump cavitation; and
- o Plenum centerline pressure and pressure gradient, fuel thermal conductivity & heat capacity.

To the extent these uncertainties apply at all to specific non-LOCA DBEs, they will be explicitly considered in the assessment of the DBE.

EXCLUDE all uncertainties based on plant variables being monitored and on definition of the transient. These include the following:

- o ~~Plenum~~ inlet temperature; and
- o ~~Assembly~~ effluent temperature.

Uncertainties in these parameters will be explicitly considered in the assessment of individual DBEs.

EXCLUDE heated wall and void effects, since limits are in terms of actual flow and

inlet temperature.

SAR CHAPTER 15 Non-LOCA ACCIDENT APPLICATION: For non-LOCA transient application, the calculated transient would be compared against the limits above, with the conclusion of no FI (and no violation of fuel design limits) if the above core limits were never exceeded. If compared in terms of inlet temperature versus assembly power, the conclusion is conservative for either increasing temperature or decreasing flow, and virtually exact for decreasing pressure. However, the amount of conservatism becomes negligible for "slow" transients with respect to the 0.5 seconds core coolant transit time. Uncertainties in the measurement of variables in the scram circuits (e.g. assembly effluent temperature) would be treated explicitly as inaccuracies in scram setpoints.

MARGIN TO DESIGN CORE SAFETY LIMITS: The safety margins could be expressed in terms of degrees inlet temperature (inlet temperature at the core safety limit less actual or calculated transient inlet temperature).

TECHNICAL SPECIFICATION APPLICATION: The "core safety limit" plots above would be translated into measured variables (assembly outlet temperature and plenum inlet temperature). Following any event, a post-mortem could determine whether the core safety limits were violated.

SAR CHAPTER 4 DISCUSSION: The basis and methodology for generation of the above safety limit would be discussed, as well as the FI treatment for LOCA.

COMMENTS ON GENERATION OF SAFETY LIMITS: The PRINCIPLES for the safety limits should be carefully reviewed; e.g., what probability level and what uncertainty factors should be included. The first recitation of these bases will set a precedent that will be difficult to change. The APPLICATION, however, should be simple and conservative. These limits will not limit restart power, and it would be wasteful to devote manpower to establishing the "least restrictive but still defensible" limit if a liberal dose of simplifying conservatism would significantly reduce manpower requirements.

APPENDIX II - MARY Code Certification Schedule

SCHEDULE FOR QUALIFICATION OF MARY REV 3 COMPUTER CODE

1. Complete Problem Statement & Solution Proposal (PSSP) 10/06/89
2. Complete training on QA Procedures 10/13/89
3. Complete Software Quality Planning Checklist 10/18/89
4. Software Certification Packages (Sections A-J are those prepared by the code author (Toby Burnett). Sections K and L are prepared by the Software Reviewer (Westinghouse Advanced Energy Systems):

Sections

	A-J	K,L
MATHSUB (Misc arithmetic routines)	09/20	11/17
LODE1 (Differential equation solver)	10/09	11/17
NEUTKIN (Point neutron kinetics)	10/09	11/22
Flows (CHANNEL, HOMONOD, PIPE)	10/18	12/15
FUEL2 (Fuel thermal kinetics)	10/20	12/15
PROPSUB (Properties of D2O & H2O)	11/10	12/21
DECAY (Decay heat)	11/15	12/21
MARSUB1 (input)	11/21	12/29
MARSUB2 (initialization)	11/21	12/29
MARSUB3 (output)	11/21	12/29
MARSUB4 (transient)	11/21	12/29
MARY (logic control)	11/30	1/9

NOTE: With the completion of the certification packages,
MARY rev 3 is "Certified" as defined by QAP IV-2.

5. Software Verification and Validation Report * 01/22/90
6. MARY configuration on mainframe 01/23/90
7. Complete Software Benchmark Testing Report * 03/30/90
(Comparisons of MARY with AA3 and plant data)

* These items are for the issuing of a report as described in item 4 as well as benchmarking activities throughout code development.

END

**DATE
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