UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W.
ATLANTA, GEORGIA 30323

Reports Nos.: 50-327/88-35 and 50-328/88-35

Licensee: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-327 and 50-328 License Nos.: DPR-77 and DPR-79

Facility Name: Sequoyah Units 1 and 2

Inspection Conducted: July 11-15 and August 22-23, 1988

Inspectors: M W Branch September 12, 1988
M. W. Branch, Team Leader Date Signed

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Approved by: Kenneth M Jenison September 12, 1988
K. M. Jenison, Acting Chief Date Signed
Projects Section 1
Division of TVA Projects

SUMMARY

Scope: This special, announced inspection reviewed conditions surrounding the recently identified Sequoyah Unit 2 excessive post trip cooldown condition and its affect on end of life shutdown margin. The inspection was conducted on site and at the corporate office to independently assess the event and to evaluate the the licensee near term and long term corrective actions.

Results: The inspection determined that the licensee had failed on numerous occasions to take effective corrective action necessary to maintain the plant as described in the FSAR, which is the reference document for many activities, including reload core design.

Specifically, the excessive cooldown following a reactor trip had been identified in 1982 as part of the startup test program. However, proper and adequate corrective action was not taken at that time. Had the licensee taken and documented effective corrective action, as required by 10 CFR Part 50 Appendix B, Criterion XVI, the current problem of reduced shutdown margin would not have occurred.

Subsequent to the initial failure to take adequate corrective action, the problem was not identified and pursued as a discrepancy during any of the numerous post-trip reviews or in the reload licensing interface process. The post-trip reviews did not adequately compare the actual plant performance with the bases in the FSAR. Through the reload licensing checklist, the licensee was specifically requested to reconfirm for each reload cycle that the FSAR RCS temperature value was still valid.

Additionally, the standard Westinghouse Generic Emergency Response Guideline procedure ES-0.1 was modified during procedure development by TVA without recognizing that the intent of the procedure was to quickly establish stable RCS temperatures at or above the no-load value to preserve shutdown margin.

The following Violations were identified:

° Failure to take adequate corrective action when the excessive cooldown discrepancy was first identified (Paragraph 4); followed by subsequent failures to identify or take adequate corrective actions during the post trip review process (Paragraph 7.a), as well as the 10CFR 50.59 core reload analysis (Paragraph 7.b), and the emergency procedure implementation process (Paragraph 7.c).

The following Unresolved Item was identified:

° Applicability of 10 CFR Part 21 requirements (Paragraph 8.b)

Completion of corrective actions to assure resolution of the excessive cooldown encroachment on shutdown margin at end of life for Unit 1 is a restart item.

Note: A list of the acronyms and abbreviations used in this report is found in Paragraph 10.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. Bynum, Vice President, Nuclear Power Production
   *G. Gault, Reactor Engineering Supervisor
   *J. Lemons, Nuclear Engineer, Nuclear Fuels
   *J. Robertson, Manager, Nuclear Fuels
   *B. Schofield, Licensing Engineer
   *S. Smith, Plant Manager

   Westinghouse Employees

   Nancy Campbell, Westinghouse Fuels Project Manager
   Noel Pogorzelski, Commercial Nuclear Fuel Division Core Engineer

   Other licensee employees contacted included engineers, operators, and office personnel.

   NRC Personnel

   D. Fieno, Section Leader, Reactor Systems Branch, NRR
   H. Richings, Engineer, Reactor Systems Branch, NRR

   *Attended exit interview

2. Inspection Objectives

   The Unit 2, Cycle 3 core is the first at Sequoyah to use low-leakage design, which has significant safety and economic benefits. However, one consequence of that design was that much of the excess shutdown margin present in the earlier cores was lost. Near the end of Cycle 3, the design shutdown margin values approached the minimum value required by TS.

   During the series of reactor trips which followed the restart of Sequoyah Unit 2, it was identified that the post-trip RCS cooldown exceeded the design average coolant temperatures presented in the FSAR and used in the accident analyses. The affect of these excessive cooldowns was to decrease the available reactor shutdown margin from the values assumed in the cycle design and safety analysis.

   The scope of this inspection was to evaluate information surrounding the discovery and proposed near and long term corrective actions associated with the adequacy of shutdown margin for Units 1 and 2, subsequent to reactor trips. The following objectives were established:

- To summarize the chronology of the events leading up to the identification of the problem, and the near term actions taken to address the problem (Paragraph 3)

- To compare the plant response in the area of RCS cooldown following reactor trips to the expected response documented in the FSAR and the PLS documents (Paragraph 4)

- To independently verify adequate shutdown margin for several of the previous reactor trips associated with the Unit 2 Cycle 3 core design (Paragraph 5)

- To review with Westinghouse personnel the assumptions and bases of the core reload analyses relative to RCS temperature, and to ensure that the margins used in the main steam line break analysis are preserved for continued plant operation (Paragraph 6)

- To evaluate the adequacy of the post-trip review process for identifying post-trip plant performance which deviates from the FSAR (Paragraph 7.a)

- To evaluate the TVA design controls and vendor interface associated with the core reload licensing (Paragraph 7.b)

- To evaluate the adequacy of the 10 CFR, Part 50.59 safety evaluations for core reloads with respect to this problem (Paragraph 7.b)

- To determine the reportability of the discovery of the shutdown margin problem under 10 CFR Parts 50.72 and 50.73 (Paragraph 8.a)

- To review the fuel service and design analysis contract to ensure that 10 CFR, Part 21 requirements are implemented (Paragraph 8.b)

- To review the proposed near term corrective actions (compensatory measures), which involve modification to the standard Westinghouse emergency procedure to instruct the operator to emergency borate if RCS temperature falls below prescribed values and to review the Westinghouse proposed modifications to the shutdown margin procedure to ensure that they are properly reflected in the TVA procedure (Paragraphs 6 and 9.a)

- To discuss with TVA long term corrective action to address plant cooldown subsequent to reactor trips (Paragraph 9.b)

3. Chronology of Events

The following is a brief chronological summary of the events and discussions which led to the discovery and the near term corrective actions for the shutdown margin problem. The chronology is based on information prepared by the licensee.

5/19/88 - Unit 2 tripped from 72 percent power and cooled to a T_{ave} of approximately 516°F.

5/20/88 - The NRC resident inspector questioned the excessive cooldown associated with the 5/19/88 reactor trip.

5/23/88 - Unit 2 tripped from 70 percent power and cooled to a T_{ave} of approximately 512°F. The SI-38 Shutdown Margin calculated following this trip ind[illegible] that adequate shutdown margin was maintained du[illegible] RCS cooldown, with an excess margin of 10 ppm [illegible] above the requirement.

6/6/88 - Unit 2 tripped from 98 perc[illegible] and cooled to T_{ave} of approximately [illegible]°F. [illegible] NRC resident inspector questioned the [illegible].

6/13/88 - The licensee initially id[illegible] issue of shutdown margin reduction for post-tr[illegible]

6/14/88 - The licensee initiated CAQR [illegible]0375 to document the shutdown margin problem, and discussed the core design with Westinghouse.

6/17/88 - Westinghouse confirmed by analysis that SDM was not violated in the 6/6/88 reactor trip.

Westinghouse transmitted the minimum allowable RCS temperature (519°F) for maintaining the required SDM following a reactor trip from 70 percent reactor power (Reference 88TV*-G-0049). The specified temperature was based on a reevaluation of the conservatisms used in the original calculations. The 70 percent power level was based on plans to extend Cycle 3 operation into January 1989.

The Reactor Fuels and Analysis Branch issued boration volumes required for the maintenance of SDM following reactor trips with cooling below 520 degrees F and an initial power level of 70 percent (L32 880617901).

6/18/88 - ES-0.1, Emergency Procedure for Reactor Trip Response, was revised to ensure compliance with technical specifications. Revision 3 incorporated a recommendation that auxiliary feedwater flow be limited to maintain RCS temperature above 520°F or a manual boration of the RCS be initiated.

6/19/88 - NRC discovers and informs plant management that PLS has language to the effect that plant cooldowns should be restricted to preserve shutdown margin. TVA then notifies

NRC residents that ES-01 had been changed to compensate for cooldown problems.

6/21/88 - PORS initiated PRO 2-88-178 for possible reporting of the shutdown margin problem.

6/27/88 - Westinghouse transmitted allowable cooldown temperatures following reactor trips from 80, 90, and 100 percent reactor power (88TV*-G-5556).

6/29/88 - Westinghouse RF&A transmitted boration volumes required for the maintenance of SDM from pretrip power levels of 70, 80, 90, and 100 percent (L32 880629 904).

At the time of the inspection, Revision 4 to ES-0.1, providing guidance for full power operation through end of cycle based on Westinghouse data (88TV*-G-0057), was in draft form. Emphasis was being placed on maintenance program activities to repair steam leaks and help mitigate the excessive cooldown. Longer term options for mitigating post trip cooldown and addressing SDM requirements were being pursued.

4. Comparison of Actual Plant Cooldown Response to Design Response

The inspectors evaluated the cooldown response of Unit 2 to several reactor trips, and compared these responses to expected system design responses contained in the FSAR. The post trip cooldown problem applied to both units. However, for purposes of this inspection, the inspectors concentrated on specific data associated with Unit 2.

The Sequoyah F ⁄ states that accident analyses of the plant are based on a no-load averag. RCS temperature of 547°F following a reactor trip. Section 7.7, entitled Control Systems, and Section 15.1, entitled Accident Analysis, Normal Operations and Operational Transients, both indicate that the control system are designed and groomed to maintain a post trip no-load T_{ave} of 547°F. Specifically, Section 7.7.1 states, "The steam dump feedwater control systems are designed to prevent the average coolant temperature from falling below the programmed no-load temperature following the trip to ensure adequate reactivity shutdown margin".

The magnitude of th. excessive RCS cooldowns for Unit 2 Cycle 3 are demonstrated by the data below, which was provided by the licensee:

Trip#	Date	Burnup	Power	Xe	Post Trip T_{ave}
48	12/29/84	45	15%	Equilibrium	510.0°F
49	1/12/85	371	99.4%	Transient	513.8°F
50	1/14/85	385	32.4%	Transient	530.0°F
51	2/15/85	1575	99.4%	Equilibrium	506.3°F
52	2/17/85	1582	30.2%	Transient	534.5°F
53	5/3/85	4401	99.5%	Equilibrium	525.5°F

Trip# (cont'd)	Date	Burnup	Power	Xe	Post Trip T_{ave}
54	5/22/85	4706	99.5%	Equilibrium	518.8°F
55,56 and 57 Non-Power Trips					
58	5/19/88	8196	72%	Transient	516.3°F
59	5/23/88	8227	70%	Transient	512.4°F
60	6/6/88	8669	97.8%	Equilibrium	527.1°F
61	6/8/88	8677	12.4%	Transient	522.3°F
62	6/9/88	8677	19.7%	Transient	511.9°F

As can be seen from the above data the plant typically cools to an average temperature of approximately 520°F following a reactor trip. The standard Westinghouse design for control systems indicate that the control systems should be able to maintain RCS temperature at or near the no-load value of 547°F.

The inspector reviewed the Sequoyah design specifics to determine if the control systems had a different control band from the standard Westinghouse design. The PLS document provided to TVA by Westinghouse during construction indicated that the T_{ave} control system could control temperature within a 4°F band. The PLS also contained Precaution #7 which indicated that RCS temperature must be monitored and if T_{ave} was not being properly controlled following a reactor trip the operator should reduce feed water in order to preserve shutdown margin.

The licensee has maintained that the plant has always experienced the magnitude of cooldowns typical of those discussed above. To establish a starting point for when these excessive cooldowns began the inspector requested and was provided a copy of the initial startup test for plant response following a trip from 100% reactor power (SU-9.4A). This test was performed in May 1982. The inspector's review of this test indicated that a test deficiency (2-9.4A-1) was written against step 6.8.3 which required that T_{ave} steady out at or above no load T_{ave} without manual intervention on feedwater flow. The test deficiency indicated that the control system could control no better than 12°F below the no-load value. The test procedure was accepted by the PORC and annotated to the effect that the deficiency was acceptable since there was no mandatory acceptance criterion which it failed to meet. Additionally, the deficiency was later reevaluated as still being acceptable based on the fact that modifications to the main feed system to require a feed pump trip whenever a feed isolation occurred were complete. However, no retests were performed. The fact that the excessive cooldowns continued after the modifications were complete, and that no actions were taken to address this discrepancy between actual plant response and the values in the FSAR, indicate that the corrective action for the problem, which was identified through testing, was ineffective. Although it was identified that plant operations were not as described in the FSAR, no written safety evaluation was performed as required by 10 CFR Part 50.59 (See paragraph 7.b). No actions were initiated to update the FSAR as required by 10 CFR Part 50.71 (e).

10 CFR, Part 50 Appendix "B" Criterion XVI, Corrective Action, requires that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to this requirement, the evaluation and correction of the test deficiency associated with the ability to control RCS temperature following a reactor trip was improper and ineffective, in that, system performance stated in the FSAR was not obtained resulting in the reduction of safety margin associated with reactor shutdown margin. This is identified as the first example of Violation 327,328/88-35-01.

5. Independent Calculation of Shutdown Margin

NRC inspectors independently calculated shutdown margin for several of the previous reactor trips on Unit 2 Cycle 3 to verify that adequate margin had been maintained.

The NRC requested the shutdown margin data for the five trips that occurred since initial Unit 2 restart. These trips were identified by the licensee as trip numbers 58 thru 62. The licensee was only able to provided data for trips 58, 59 and 60 prior to the completion of the inspection, as they indicated data for the other trips were in the duplication process.

Using the licensee's procedure (SI-38), the inspectors verified the shutdown margin calculations of the licensee. The inspectors' calculations were in general agreement with the licensee results. The inspectors then calculated the shutdown margin for trip number 60, which occurred on June 6, 1988 from 98 percent power, using the worst case RCS temperature (the lowest RCS temperature reached during the transient). The results of this calculation indicated that, by using the worst case temperature and the revision of SI-38 in effect at the time of the trip, the TS required shutdown margin of 1.6 % delta k/k was not maintained. This calculation indicated that the boron concentration required to preserve the TS SDM requirement would be 497.7 ppm, whereas the actual boron concentration was 442 ppm. However, as stated in the chronology listed earlier, the licensee had Westinghouse perform a more refined and precise calculation of the actual shutdown margin at the lowest RCS temperature. The results of this calculation, transmitted 6/17/88 (88*-G-0049), indicated that the actual shutdown margin was greater than that required by TS.

The licensee was requested to calculate, using the lowest RCS temperatures reached, the shutdown margins for trips 61 and 62. Both of these calculations indicated that the TS shutdown margin values were met.

Thus, the inspectors confirmed that adequate shutdown margin had been maintained throughout Unit 2 cycle 3 operation.

It should be noted that the surveillance requirements for TS 3.1.1.2 do not require an immediate determination of shutdown margin following a reactor trip and in no case does the TS require the lowest RCS temperature be used. However, the TS requires that the shutdown margin be preserved at all times. Additionally, the Westinghouse developed emergency procedure for actions required after a reactor trip requires that a shutdown margin calculation be performed as supplemental action. The licensee's position is that they perform the shutdown margin calculation for the actual conditions that exist at the time of performance and that this method is consistent with other utilities that were polled.

6. Discussions of Shutdown Margin Design Assumptions with Westinghouse

The inspectors reviewed with Westinghouse personnel the assumptions and bases for the Sequoyah reload analyses to determine whether adequate shutdown margin exists on Unit 2 through the remainder of the current cycle.

The Westinghouse design and analysis provided for the 1.6% delta k/k shutdown margin required by TS to be preserved at end of life, when the core was most vulnerable to the steam-line-break accident. The FSAR for Sequoyah described plant behavior post-trip as a cooldown to the no-load average coolant temperature of 547° F. The Westinghouse analysis accounted for instrument errors by assuming the plant was operating 2 degrees F above full-load average coolant temperature immediately prior to the trip and cooled to 2 degrees below no-load average temperature following the trip, but the analysis did not provide any margin for any actual cooldown below 547°F.

Once Westinghouse was informed that the actual post-trip cooldowns were excessive, to as low as 510° F in one loop, they performed a bounding calculation. That calculation confirmed that none of the cycle 3 trips and cooldowns through June 6, 1988 had reduced shutdown margin below the limit. However, calculations for EOC conditions, when the moderator temperature coefficient is most negative, showed that under some conditions trips from 100% RTP would lead to insufficient shutdown margin if the cooldown was to 544 degrees F or less. Even trips from 100% RTP at nominal conditions, equilibrium xenon and D bank above 200 step withdrawn, would lead to reduction in shutdown margin if the RCS cooled to less than 532° F. Since the cooldown for recent trips has been about 520° F, it was clear that additional action was necessary.

Past practice by Westinghouse has been to reduce calculated control rod worth at any core condition by 10% to account for observed differences between predicted and measured control rod worths at BOL. However, Westinghouse has justified, per approved topical report WCAP 9217, a reduction of the factor of conservatism to 7%. That recalculation of control rod worth would provide some additional shutdown margin. However, FSAR Table 4.3.2-3 currently shows shutdown margin to be calculated using 10% calculated rod worth reduction. Discussions with NRR Reactor Systems Branch personnel revealed that they had no technical reservations about reducing the rod worth adjustment to 7%. However, both they and the inspectors objected to a further proposal by Westinghouse to reduce the adjustment down to 5% for the current cycle of Unit 2. The basis for that proposal was that the BOL measurement of total control rod worth was 2% greater than calculated. However, that test confirms the adequacy of the rod-worth-calculation model at BOL, it does not guarantee that the EOC rod-worth calculation would also be an underestimate.

At the time of the inspection, Unit 2 was being operated at 70% RTP both to extend core life and to reduce the cooldown encroachment on shutdown margin. Worst case, rods at the insertion limit and transient xenon conditions, calculations by Westinghouse showed that a post-trip cooldown from 70% RTP to 519° F is acceptable. NRC Generic Letter 84-21 addressed the possible adverse effects of "LONG TERM LOW POWER OPERATION OF PRESSURIZED WATER REACTORS." Westinghouse personnel stated that specific guidance on that concern had been provided to TVA, and that the plant was currently operating in conformance with the guidance. The essential element of the guidance is to maintain the axial power distribution at reduced power consistent with that expected at full power. That is accomplished by operating with control rods partially inserted at reduced power.

7. Evaluation of Design Controls and Vendor Interface

After the excessive cooldown was identified as a test deficiency in 1982, but was not adequately corrected, there were a number of additional opportunities to identify the problem. The inspectors reviewed activities related to design and safety review processes to determine why the excessive cooldown and the resulting effect on shutdown margin were not identified through other available mechanisms.

a. Post Trip Review Process

The inspectors questioned why the discrepancy between the actual RCS temperature and the FSAR values following a reactor trip was not identified during any of the post-trip reviews which had been performed.

The inspectors reviewed the post trip review procedure AI-18, File 18 in order to determine if there was a requirement to evaluate core performance (i.e., shutdown margin) for the lowest temperature reached during the transient. This procedure does not specifically require that actual core or plant performance be evaluated against specific FSAR transient analysis requirements. It has only a simple (yes/no) statement that all designated parameters were within expected limits. Had the post trip procedure been more detailed and required the evaluation of worst case shutdown margin or compared actual post trip parameters with FSAR values, the fact that control systems were incapable of satisfying FSAR requirements regarding lowest T_{ave} following a reactor trip and the calculated violation of shutdown margin requirements would have been identified earlier. The licensee was requested to evaluate the procedure for calculating post-trip shutdown margin and determine if required results can be achieved with current procedure detail.

The purpose of the post trip review is to identify and correct conditions that are not acceptable and affect plant expected response to trip conditions. The failure of recent post trips reviews performed subsequent to the May 19 and 23, 1988 and June 6, 1988 reactor trips as well as post trip reviews performed prior to the August 1985 shutdown, to identify and correct the plant cooldown condition is identified as the second example of violation 327, 328/88-35-01 for ineffective corrective action .

b. Vendor Interface and 50.59 Safety Evaluation for Reload Cycles

To determine why the cooldown discrepancy was not uncovered during the design review for the core reloads, the inspectors examined the interface between the licensee and the vendor, and reviewed the 10 CFR 50.59 safety evaluation performed for Unit 2 Cycle 3 and Unit 1 Cycle 4.

Although not totally within the scope of this inspection, the Westinghouse interface efforts supporting a core reload analysis were briefly discussed with the Westinghouse Fuel Project Manager and a Westinghouse core engineer. The Westinghouse personnel presented the following scope and timetable for the reload design interface process:

FORMAL TVA/WESTINGHOUSE COMMUNICATIONS

RELOAD DESIGN PROCESS

	MONTHS PRIOR TO STARTUP
A. INITIALIZATION PHASE	
° Reload Safety and Licensing Checklist	18
° Energy Requirements Preliminary Loading Pattern	16
° Design Initialization Meeting	14
° Design Schedule	13
B. CORE MANAGEMENT PHASE	
° Loading Pattern Established	12

MONTHS PRIOR TO STARTUP (cont'd)

C. SAFETY ANALYSIS PHASE

° Reload Safety Analysis Checklist	6
° Reload Safety Evaluation	4

D. OPERATIONS INFORMATION

° Nuclear Design Report and Operations Data	1

The reload safety and licensing checklist was described by Westinghouse as the vehicle where plant specific performance parameters are transmitted between Westinghouse and TVA for the purpose of validation of data. The following is an excerpt from the Sequoyah Unit 2 Cycle 3 Westinghouse Reload Safety and Licensing Checklist, Revision 0, dated 7/21/83.

> The performance characteristics of plant components or safety systems assumed in prior safety analyses are important input to the safety analyses for the next cycle. Unless otherwise advised, the performance characteristics found in the following documents will be assumed for the next cycle's licensing effort. The documents revisions must be consistent with the date of issuance of the completed checklist.
>
> (1) The original Plant Safety Analysis Report
>
> (2) Loss of Coolant Accident Submittals
>
> (3) Fuel densification submittals (if any)
>
> (4) Reload Safety Evaluation Reports
>
> (5) Technical Specifications (approved or submitted)
>
> (6) Any other special analyses such as Anticipated Transient Without Trip (ATWT) analyses, analyses of the effect of a modified system, etc. unless addressed in your response, Westinghouse must assume that the only changes in core characteristics for the reload are those found in the design of the reload core.
>
> Section II-C Thermal Hydraulics
>
> (a) Change in operating pressure - none
> (b) Change in operation temperature - none
>
> T in --- 547 degrees F
> T ave --- 547 degrees F to 578 degrees F
> Delta T --- 60.3 degrees F

Thus, the Westinghouse reload licensing checklist specifically states that the FSAR values, including no-load T_{AVE}, will be assumed in the reload design analysis unless Westinghouse is notified otherwise by TVA. Therefore, the licensee had been formally made aware of the continuing use of the originally assumed no-load T_{AVE} design value for the reload calculations.

The inspector evaluated the adequacy of the TVA 10 CFR Part 50.59 evaluations for the Unit 2 Cycle 3 and Unit 1 Cycle 4 reloads to attempt to determine why the expected effects of excessive plant cooldowns were not addressed in these safety evaluations. USQD 84-34 for Unit 2, dated 9/7/84, and USQD 85-20 for Unit 1, dated 11/1/85, contained only 2 pages and consisted of nothing more than cover sheets with signature blocks for the Westinghouse Reload Safety Evaluation.

10 CFR 50.59 allows the holder of a license to make changes in the facility as described in the safety analysis report without prior commission approval unless the proposed change involves a change in the technical specifications incorporated in the licensee or an unreviewed safety question. A proposed change shall be deemed to involve an unreviewed safety question; 1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR be increased; or 2) if the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created; or 3) if the margin of safety as defined in the basis for any technical specification is reduced.

In performing the 10 CFR 50.59 safety evaluations for the reload cores, the accuracy of the post-trip cooldown values presented in the FSAR were assumed to be correct and were not questioned, based on the assumption that the FSAR had been kept up to date.

The initial failure to take adequate corrective action when the excessive cooldown was initially identified including the failure to comply with 10 CFR 50.59 at that time was addressed in paragraph 4 as violation 327,328/88-35-01. As a result of the initial failure, both Units 1 and 2 had been operated since licensing outside the system design described in the FSAR. Specifically, FSAR Section 7.7.1 required that the steam dump and feed water control system be designed to prevent the average coolant temperature from falling below the program no-load temperature following a reactor trip to ensure adequate reactivity shutdown margin is preserved. The excessive cooldown constituted a change to the operation of the facility as described in the FSAR and should have been supported by a written safety evaluation.

The Westinghouse analysis, based on the criteria established by the Reload Safety and Licensing checklist discussed above, used as a basis a post trip T_{ave} value of two degrees less than the no-load value of 547°F (i.e. 545°F). Using this post trip temperature of 545°F resulted in a calculated EOC shutdown margin of 1.61 % delta k/k compared to the TS required value of 1.6% delta k/k for Unit 2 and 1.64 % delta k/k compared to the TS value of 1.6 % delta k/k for Unit 1. Although the amount of excess shutdown margin available at EOC tended to be lessened by the low leakage design, this reduction in itself did not constitute an unreviewed safety question per 10 CFR 50.59 as long as the required 1.6 % delta k/k was maintained.

Review of correspondence from Westinghouse to TVA regarding this issue included a June 27, 1988 letter (88TV*-G-0056) in which Westinghouse stated that cooldown temperatures as low as 520°F might result in future loading pattern restrictions, which would reduce the low-leakage capability, with loss of its attendant advantages. The amount of shutdown margin reduction associated with a post-trip temperature change from 545°F to 520°F would reduce the EOC shutdown margin value by approximately 0.7% delta k/k down to approximately 0.9 % delta k/k, which is below the allowed TS value of 1.6% delta k/k. Westinghouse recommended limiting the cooldown as discussed in paragraph 7c.

The adequacy of the reload cycle 10 CFR 50.59 evaluations, which did not identify as an unreviewed safety question the decrease in EOC shutdown margin below the value allowed by TS is identified as the third example of violation 327,328/88-35-01 for ineffective corrective action.

The inspector reviewed the following Nuclear Fuels Procedures to determine if, in general, adequate design and interface controls exist:

- NFP 7.0, Control of Reload Core Design and Analysis
- NFP 7.1, Organization and Interface for Reload Design and Analysis
- NFP 7.2, Reload Design Document Control

The inspector concluded, based on the procedures reviewed, that in general, adequate controls did exist to obtain a proper core reload analysis. However, the procedures reviewed were issued in 1987 rather than 1983 when the analysis was performed. The manager of Nuclear Fuels indicated that similar procedures did exist at the time the core reload analysis was performed. Additionally, it should be noted that even the current procedures will be subsequently modified to reflect the February 1988, organizational change that made the Nuclear Fuels Division a part of DNE.

c. Emergency Procedure Review

Since part of the TVA corrective action was to modify the standard Westinghouse owner's group emergency operation procedure to compensate for the excessive cooldowns, the inspector conducted a review of the procedure and the Westinghouse guidance. Specifically, procedure ES-0.1 Reactor Trip Response was reviewed. The Westinghouse guideline contained language to the effect that RCS and secondary plant stabilization at no-load conditions was part of the procedures major action goals. In fact, the logic tree for step 1 of the procedure shows actions required for temperature decreasing below the no-load values as stop dumping steam followed by controlling AFW flow to maintain SG level at the bottom of the level band and to isolate the main steam line if necessary.

The TVA implementing procedure issued October 4, 1984, did not specify the Westinghouse course of action to preserve temperature at or above the no-load value. The TVA procedure stated that if Tave is decreasing in an uncontrolled manner, then verify steam dumps and secondary PORV closed followed by closing the MSIVs and their bypass valves. This method does not appear to preserve the no-load temperature and consequently the reactor shutdown margin. The inspector reviewed the TVA step deviation for this procedure issued subsequent to procedure implementation and determined that the TVA basis for this deviation was that the AFW system design includes automatic level control valves and therefor manual control of AFW is not necessary. This deviation does not appear to address the issue at hand to preserve shutdown margin possibly at the expense of reducing AFW flow to the SGs.

If, at the time of implementation of the generic guides in October 1984, TVA had questioned the purpose of the steps in the Westinghouse procedure the excessive cooldown/shutdown margin problem may have been properly resolved at that time. This failure to identify and correct a nonconforming condition is identified as the fourth example of violation 327,328/88-35-01 for ineffective corrective action.

8. Reportability

a. 10 CFR 50.72 and 50.73

As indicated above, the NRC considers that the reduction in EOC shutdown margin associated with the excessive plant cooldown constituted an unreviewed safety question and could have resulted in the plant being in a condition that was outside the design basis. In fact, the licensee's near term corrective action was to limit reactor power to 70 percent and to change the standard Westinghouse post-trip emergency procedure as a compensatory measure to ensure that the plant could be operated within the design basis.

The CAQR (SQP880375) dated 6/14/88 indicated that the discovered condition was not reportable. The copy of PRO (2-88-178) dated 6/21/88 provided to the inspectors did not have a reportability determination made at the time of the inspection. However, the licensee did provide the written report , LER 328-88-030 within the required 30 day period.

b. 10 CFR Part 21

Due to the potential generic implication of the above shutdown margin problem, the inspector reviewed the Fuels and Analysis service contract (68p-84-T1) between TVA and Westinghouse to determine if the requirements of 10 CFR 21 regarding vendor responsibility as to notification were applicable. The inspector determined that the original contract dated in 1968 was issued prior to the January 6, 1978, date specified in 10 CFR 21. However, this contract has been amended several times since Part 21 first became applicable. None of the contract amendments contained language that the requirements of 10 CFR Part 21 apply. The contract did however, contain language to the effect that all NRC rules and regulations both current and future apply. The inspector requested that the licensee evaluate the current contract and determine if it should be amended to specifically state that Part 21 applies. This item is identified as unresolved item 327,328/88-35-02 pending further NRC review with the licensee and the NRC vendor branch.

9. Review of Corrective Measures

a. Near Term Compensatory Measures

As previously described, the licensee's near term corrective actions included operating at a reduced power level of 70% RTP with power distribution guidance provided by Westinghouse.

In addition, current and proposed plant procedures require post-trip emergency boration as a compensatory action to restore shutdown margin rapidly if the cooldown is beyond power and burnup dependent limits. The limits and required boration were obtained from Westinghouse, but before they were accepted and implemented they were subjected to independent review and analysis by the TVA PWR Core Design Section of the Reactor Fuel and Analysis Department. The inspectors' review of the records confirmed that TVA used independent core performance calculations to confirm that the vendor calculations gave results equivalent or conservative with respect to theirs. The TVA methods have not been described in a topical report approved by NRR, but were deemed acceptable for quality control purposes. Finally, the TVA calculations were reviewed by independent reviewers and the differences from Westinghouse results rationalized by a reviewer familiar with Westinghouse methodology. The TVA staff generated curves and tables of required boration as a function of burnup, power level, and cooldown using a computer program written in house. That program is well-documented internally, and has been accepted by peer review.

The inspectors concluded the TVA review of both Westinghouse and internal calculations was satisfactory in both conduct and documentation. The inspectors did express one concern with the procedures that have or will result from these activities. The procedures will specify the volume of 21,000 ppm boric acid to be injected. At other facilities the boric acid and primary water flow integrators have not shown acceptable accuracy for this purpose. The calibration and reliability of the boric acid integrator was not established during this inspection. The inspectors expressed this concern to the TVA staff during the inspection and to management at the exit interview on July 14, 1988.

The inspectors discussed the licensee's calculation activities and proposed compensatory actions and procedures with members of the NRR Reactor Systems Branch. The NRR staff had no criticism of either the calculations or compensatory action for Unit 2 as described to them by the inspectors. The NRR staff did express reservations about accepting similar compensatory action for Unit 1, which is faced with the same problem at EOC, but has yet to restart after being refueled during the current outage. That reservation was forwarded to plant management at the exit interview. Management stated they did not intend to restart Unit 1 until the basic problem of excessive cool-down to an unanalyzed temperature had been resolved. Management further stated they intended to complete their determination of the best method to limit post-trip cooldown within 30 days.

b. Long Term Corrective Actions

As part of this inspection the inspectors planned to discuss details of TVA's plans to minimize RCS cooldowns following reactor trips. The licensee indicated that they are currently investigating several methods to attempt to control cooldowns. They include a change in steam dump setting from T_{ave} control to Steam Pressure control, and possible modifications to the auto-level controls associated with the AFW system. Details and schedule were not discussed.

10. List of Acronyms and Abbreviations

AI - Administrative Instruction
AFW - Auxiliary Feedwater
ATWT - Anticipated Transient without Trip
CAQR - Condition Adverse to Quality Report
CFR - Code of Federal Regulations
BOC - Beginning of Cycle
BOL - Beginning of Life
DNE - Division of Nuclear Energy
EOC - End of Cycle
ES - Emergency Procedure
FSAR - Final Safety Analysis Report
LER - Licensee Event Report
MSIVs - Main Steam Isolation Valves
NFP - Nuclear Fuels Procedure
NRC - Nuclear Regulatory Commission
NRR - Nuclear Reactor Regulation
PLS - Precautions Limitations and Setpoint Document
PORC - Plant Operations Review Committee
PORS - Plant Operations Review Staff
PORV - Power Operated Relief Valve
PPM - Parts Per Million

PWR - Pressurized Water Reactor
RCS - Reactor Coolant System
RF&A - Reactor Fuels and Analysis
RTP - Rated Thermal Power
SAR - Safety Analysis Report
SDM - Shutdown Margin
SG - Steam Generator
SI - Surveillance Instruction
SQN - Sequoyah
SU - Start Up Test Procedure
T_{AVE} - Average Reactor Coolant Temperature
TS - Technical Specifications
TVA - Tennessee Valley Authority
URI - NRC Unresolved Item
USQ - Unreviewed Safety Question
USQD - Unreviewed Safety Question Determination
XE - Xenon

11. Exit Interview

The inspection scope and findings were summarized on July 14, 1988, and again on August 23, 1988, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. During the course of the inspection the inspectors were provided numerous documents which the licensee considered as proprietary. However, no proprietary material is contained in this report. Dissenting comments were not received from the licensee during the July 14, 1984 exit. However, during the August 23, 1988 reexit the licensee did comment that their position was that the shutdown margin problem was licensee identified and was not prompted by the NRC questioning of the excessive cooldown discussed in this report.

Item Number	Description and Reference
327,328/88-35-01	Violation: Failure to take adequate corrective action when the excessive cooldown discrepancy was first identified (Paragraph 4); followed by subsequent failures to identify or take adequate corrective during the post trip review process (Paragraph 7.a), as well as the 10 CFR 50.59 core reload analysis (Paragraph 7.b) and the emergency procedure implementation process (Paragraph 7.c).
327,328/88-35-02	Unresolved Item: Determine the applicability of 10 CFR Part 21 requirements (Paragraph 8.b)