Docket No. 50-213
B13208

Attachment 1

Haddam Neck Plant

Proposed Changes to Technical Specifications
Cycle 15 Coastdown Operation

April 1989

VENTING

1.39 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a VENTING process.

END-OF-CORE LIFE

1.40 END-OF-CORE LIFE shall correspond to a reactor operating condition with all control rod banks fully withdrawn, essentially 0 ppm boron concentration in the reactor coolant, and the Reactor Coolant System average temperature (Tavg) no longer maintained at normal operating temperature or the normal rated thermal power (RTP) no longer maintained.

3.17 POWER DISTRIBUTION LIMITS

3.17.1 AXIAL OFFSET

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.17.1.1 The AXIAL OFFSET shall be maintained within the limits of Figures 3.17-1 a, b, or c.

APPLICABILITY: MODE 1, ABOVE 40% of RATED THERMAL POWER.

ACTION:

With the AXIAL OFFSET outside the Acceptable Operation Limits specific in the above figures, within 15 minutes initiate corrective action and continue the corrective action so that the Axial OFFSET is within limits within 2 hours or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

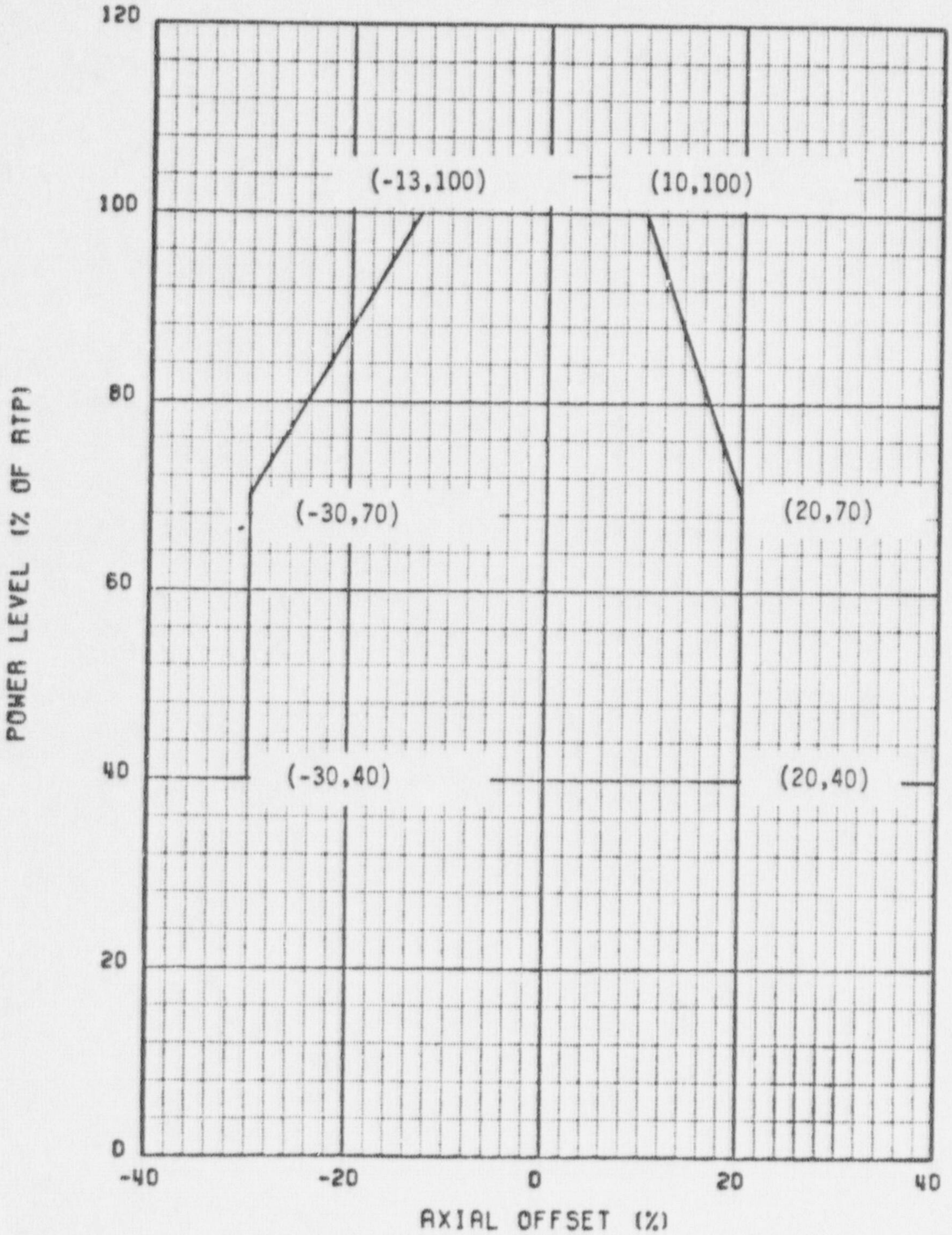
a. The AXIAL OFFSET shall be determined to be within the Acceptable Operation Limits of Figures 3.17-1a, b, or c by monitoring the AXIAL OFFSET using at least two OPERABLE! excore Power Range channels and applying the excore/incore correlation on a continuous basis.

b. The excore/incore correlation shall be verified at least once per 31 EFPD and adjusted at least once per 92 EFPD using the results of the measurements obtained in accordance with Specification 3.17.2.

c. The excore/incore correlation shall be determined after each fuel loading or major change in excore Power Range instrumentation prior to exceeding 80% of RATED THERMAL POWER.

d. The excore Power Range detectors shall be calibrated/correlated relative to the Movable Incore Detector System measurements within 7 days after completion of incore measurements.

FIGURE 3.17-IC: POWER LEVEL VS. AXIAL OFFSET LIMITS, CYCLE 15 COASTDOWN, FOUR LOOP OPERATION



POWER DISTRIBUTION LIMITS

3.17.2 LINEAR HEAT GENERATION RATE

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.17.2.1 All LINEAR HEAT GENERATION RATES (LHGRs) shall not exceed the following kilowatt per foot limits for cycle residency time:

a.	Less than 125 EFPD	13.3 kW/ft
b.	125 To 250 EFPD	13.3 kW/ft
c.	Greater Than 250 EFPD But Less Than END-OF-CORE LIFE	14.6 kW/ft
d.	Greater than END-OF-CORE LIFE During Coastdown	13.5 KW/ft

APPLICABILITY: MODE 1, above 40% of RATED THERMAL POWER.

ACTION: With the LHGR of any fuel rod exceeding the limits specified above, initiate corrective action within 15 minutes and continue corrective action so that the LHGR is within the limits within 2 hours or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

1. LHGR's shall be determined to be within the above limits by a core power distribution measurement using the Moveable Incore Detector System and in consideration of the factors listed in 2 below:

   a. At least once per 31 EFPD,

   b. Prior to THERMAL POWER exceeding 80% of RATED THERMAL POWER after each fuel loading, and

   c. After reaching 100% of RATED THERMAL POWER and achieving equilibrium xenon conditions after each refueling.

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Attachment 2

Haddam Neck Plant

Technical Report Supporting Cycle 15 Operation
Coastdown Addendum

April 1989

NUSCO-155, Addendum
April, 1989

CONNECTICUT YANKEE ATOMIC POWER COMPANY
HADDAM NECK PLANT

Technical Report Supporting Cycle 15 Operation
Coastdown Addendum

Northeast Utilities
P.O. Box 270
Hartford, Connecticut

TABLE OF CONTENTS

List of Figures

1. Introduction and Summary

The objective of this report is to support four loop, coastdown operation of the Haddam Neck Plant at the end of Cycle 15.

The nominal end of Cycle 15 will occur at a burnup of 12000 MWD/MTU. A coastdown to a cycle burnup of 13000 MWD/MTU has been evaluated. The reviews of fuel mechanical performance in Section 4, the thermal hydraulic performance in Section 6 and the accident and transient analysis in Section 7 were based on the 13000 MWD/MTU cycle burnup for the range of expected coastdown operating conditions.

Based on the original Cycle 15 analyses (Reference 1), analyses performed for coastdown conditions and the proposed revision to Technical Specifications, it is concluded that the Haddam Neck Plant can be safely operated in the four loop, coastdown mode to a cycle burnup of 13000 MWD/MTU.

2. Operating History

Initial criticality for Cycle 15 occurred on March 19, 1988. The plant phased online March 26, 1988 and reached 100% power on April 9, 1988. The plant was shutdown for 28 days in May 1988 for maintenance, and has since operated at a thermal capacity factor of 99% through February 1989. No operating anomalies have occurred during the cycle that would adversely affect fuel performance.

3. General Description

The description of the Haddam Neck reactor core provided in Reference 1 is unaffected by coastdown operation.

4. Fuel System Design

The fuel system design of the Haddam Neck Plant is not affected by coastdown operation. However, subsequent to the Reference 1 submittal, a Post Irradiation Examination (PIE) was performed on the four Zircaloy clad Lead Test Assemblies (LTAs) during the 1987 refueling outage prior to Cycle 15 startup. This inspection showed that there was insufficient gap between the top of the fuel rods and the bottom of the top nozzle to accomodate the projected Cycle 15 burnup. Modified upper nozzles were installed to yield a gap that can accomodate a Cycle 15 burnup greater than 13000 MWD/MTU.

The mechanical evaluation of the stainless steel and Zircaloy clad fuel rods was originally performed in Reference 1, and was re-evaluated to account for the extended burnup due to coastdown operation. The results of this re-evaluation are provided below:

Cladding Collapse

The Cycle 15 power history was re-evaluated to include coastdown operation. The radial and axial power history, and therefore the cladding collapse results provided in Reference 1 bound coastdown operation for the four Zircaloy clad LTAs and all stainless steel clad fuel batches except Batch 16. The Batch 16 radial power peak at the end of coastdown is less than 2% higher than the limiting value assumed in the original Cycle 15 analysis, but this is more than offset by the 8% reduction in the axial power peak at the end of coastdown. Therefore, the cladding collapse results for Batch 16 provided in the original Cycle 15 analysis bounds coastdown operation.

Cladding Stress

The key parameters in the stress evaluation that are potentially affected by coastdown operation are:

- Maximum system pressure
- Maximum Linear Heat Generation Rate (LHGR)
- Cladding oxidation (residence time)

The system pressure is unaffected by coastdown operation. The actual and maximum LHGR decrease during coastdown. The original Cycle 15 cladding oxidation was determined by assuming a residence time of 1456 Effective Full Power Days (EFPD), compared with an actual residence time of 1203 EFPD at the end of coastdown. The minimum margins for stress intensity provided in Reference 1 bound coastdown operation.

Cladding Strain

The cladding strain analysis is primarily affected by total burnup and LHGR. The assumptions used in the original Cycle 15 analysis bound coastdown operation.

Cladding Fatigue

Fatigue usage factors are primarily dependent on fuel residence time. The original Cycle 15 analysis assumed a residence time of 1456 EFPD. The maximum residence time at the end of coastdown will be 1203 EFPD. Therefore, the original Cycle 15 results bound coastdown operation.

The Zircaloy fuel rod growth has been re-evaluated to account for the additional burnup due to coastdown. The modified upper nozzles of the LTAs yield a peak rod burnup margin of over 4000 MWD/MTU at the end of coastdown using a conservative growth model. Therefore, there is sufficient fuel rod/upper nozzle gap to accomodate the increased burnup due to coastdown operation.

The original Cycle 15 maximum fuel rod internal pressure analysis was reviewed to account for the extended burnup due to coastdown operation. The power history and maximum LHGR assumed in the original analysis bound coastdown operation. The maximum fuel rod internal pressure, therefore, will remain below nominal system pressure during coastdown operation.

5. Nuclear Design

The nuclear design parameters that are input to the safety analysis were evaluated for coastdown conditions at the burnup limit of 13000 MWD/MTU. All of the parameters provided in Reference 1 bound coastdown operation, with the exception of the maximum differential rod worth at subcritical conditions. The maximum differential rod worth has increased from 135 pcm/inch to 136 pcm/inch. Additional nuclear design parameters were also developed for the steamline break and Rod Cluster Control Assembly (RCCA) ejection accident for the range of coastdown conditions. The impact of these revised parameters on the accident and transient analysis is addressed in Section 7.

The limiting Linear Heat Generation Rate (LHGR) that resulted from a re-analysis of the Large Break LOCA for coastdown conditions (Section 7), requires a new set of axial offset operating limits for coastdown operation. The axial offset limits were determined using the Westinghouse methodology that has been reviewed and approved by the NRC.

6. Thermal Hydraulic Design

The thermal hydraulic design was re-evaluated for coastdown operation. The original Cycle 15 steady state minimum DNBR and maximum fuel temperature bound coastdown operation since the enthalpy rise hot channel factor remains bounding and the maximum linear heat generation rate decreases with coastdown.

7. Accident and Transient Analysis

The non-LOCA design basis re-analysis has been accepted by the NRC (Reference 2) as a result of the Reference 1 submittal. The impact of coastdown operation on the design basis accidents for four loop operation was assessed by determining the affected nuclear design parameters (Section 5) and evaluating the impact of the changes for coastdown conditions. The revised nuclear design parameters are bounded for all non-LOCA design basis accidents except for the uncontrolled rod withdrawal from subcritical. The steamline break and Rod Cluster Control Assembly (RCCA) ejection accidents were also re-evaluated to address the unique operating conditions during coastdown. An assessment of each of these accidents is provided below:

Uncontrolled Rod Withdrawal from Subcritical

The additional burnup due to coastdown operation has increased the maximum reactivity insertion rate from 135 pcm/inch to 136 pcm/inch. This change has a negligible impact on the minimum DNBR since the peak heat flux remains a small fraction (<15%) of the full power heat flux.

Rod Cluster Control Assembly (RCCA) Ejection

The RCCA ejection accident was re-evaluated to determine the impact of new nuclear design parameters for the range of coastdown operating conditions. The resulting fuel pellet enthalpy, average cladding temperature at the hot spot and radiological consequences remain bounded by the original Cycle 15 analysis. The peak RCS pressure, however, is significantly higher due to the lower initial RCS temperature and steam generator pressure during coastdown operation. The lower temperature and pressure combine to delay the actuation of the Main Steam Safety Valves, thus causing the RCS to pressurize to the pressurizer safety valve set point (Figure 7-1). The peak RCS pressure remains less than the value which would cause stresses to exceed the faulted condition for stress limits.

Steamline Break

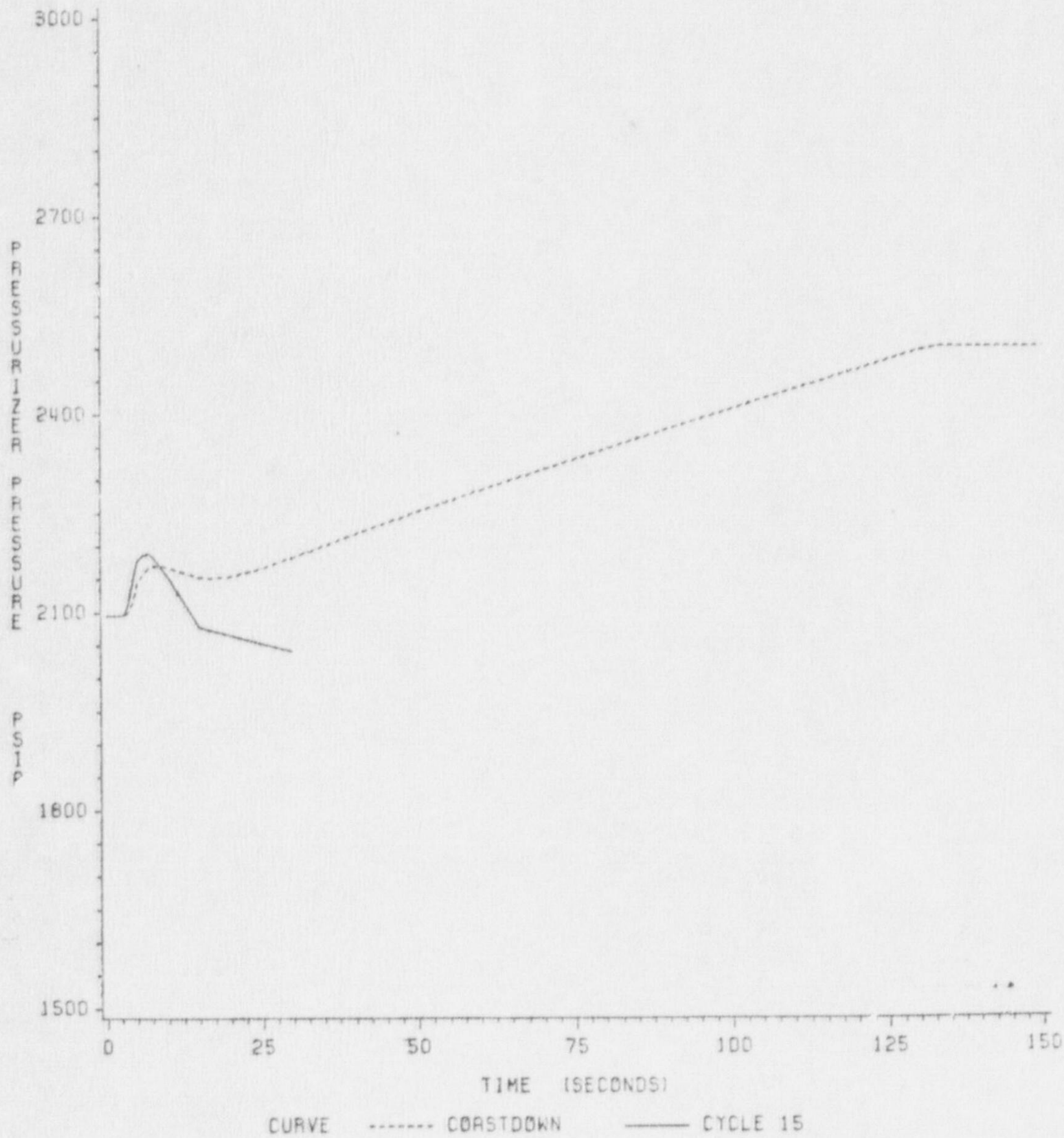
The steamline break accident was also re-evaluated to account for the range of coastdown operating conditions. The combination of initial lower reactor power and RCS temperature resulted in a slightly higher core fission power, but reduced power peaking factors used in the reactor physics state point evaluation during the accident. The net effect of these changes is that the minimum DNBR and maximum centerline fuel temperature results from the original Cycle 15 analysis remain bounding for coastdown operation.

The Small Break LOCA design basis re-analysis was approved by the NRC (Reference 3) subsequent to the Reference 1 submittal. The impact of reduced reactor power and RCS temperature during coastdown operation has been evaluated and there is no negative impact on the peak cladding temperature.

The Large Break LOCA design basis was re-analyzed to assess the impact of the reduced RCS temperature during coastdown operation. Previous coastdown sensitivity studies of the Westinghouse Interim Acceptance Criteria (IAC) evaluation model have shown that at 100% power, a reduction in the core inlet temperature yields a higher Peak Cladding Temperature (PCT). The sensitivity to the core inlet temperature, however, changes between 90% and 100% power , such that below 90% power, the effect of core inlet temperature on PCT is insignificant.

The current limiting case, which establishes the Linear Heat Generation Rate Technical Specification limits, was re-analyzed for bounding coastdown conditions. Reactor power was conservatively maintained at 102%, but the core inlet temperature was reduced from 536F to 510F. The 510F core inlet temperature corresponds to a coastdown power level of 90%. The analysis also assumed a conservative reduction in the normal end of cycle LHGR from 14.6 to 13.5 kw/ft. The resulting PCT at 13.5 kw/ft is 2213F, which yields over 80F margin to the IAC limit of 2300F.

Figure 7-1 RCCA Ejection - Coastdown Operation



8. References

1. E. J. Mroczka (CYAPCO) to USNRC, Cycle 15 Reload-Technical Specification Change Requests and Reload Report, June 1, 1987. This letter transmitted NUSCO-155, Technical Report Supporting Cycle 15 Operation, June, 1987.

2. A. B. Wang (USNRC) to E. J. Mroczka (CYAPCO), Safety Evaluation of Northeast Utilities Topical Report 151, Haddam Neck Non-LOCA Transient Analysis, Octoger 18, 1988.

3. A. B. Wang (USNRC) to E. J. Mroczka (CYAPCO), Safety Evaluation for NULAP5 Code and Its Use in Haddam Neck Small Break LOCA Analyses (NUREG-0737 Items II.K.3.30 and II.K.3.31), August 3, 1988.