Certified By:
Dana Powers - 6/14/96

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REACTOR FUELS SUBCOMMITTEE MEETING MINUTES
APRIL 10, 1996
ROCKVILLE, MARYLAND

Introduction

The ACRS Subcommittee on Reactor Fuels held a meeting on April 10, 1996 in Room T-2B3, 11545 Rockville Pike, Rockville, Maryland. The purpose of this meeting was for the Subcommittee to discuss the NRC Office of Research activities related to high burnup fuels, recent experimental results from foreign facilities for reactivity initiated accidents with high burnup fuels, and related matters. The meeting was open to public attendance. Dr. Medhat El-Zeftawy was the cognizant ACRS staff engineer and Designated Federal Official for this meeting. There were no written comments or requests or for time to make oral statements received from members of the public. The meeting was convened by the Subcommittee Chairman at 8:30 am, and adjourned at 5:15 pm.

Attendees

Principal meeting attendees included the following:

ACRS

D. Powers, Chairman
M. Fontana, Member
T. Kress, Member
R. Seale, Member
W. Shack, Member
M. El-Zeftawy, Staff
A. Cronenberg, Fellow

NRC

R. Meyer, RES
T. King, RES
D. Ebert, RES
H. Scott, RES
R. Jones, RES
L. Phillips, NRR
E. Kendrick, NRR
Y. Chen, RES

Others

T. Fujishiro, Japan
F. Schmitz, France
G. Wu, NEI
O. Ozer, EPRI
G. Potts, GE
H. Culet, Siemens

Others (Cont'd)

S. Schultz, Yankee Atomic
J. Rashid, AnaTech
N. Waeckel, EdF
M. Champ, France (IPSN)
T. Zama, TEPCO
A. Kishi, TEPCO
K. Oshima, Toshiba
R. Anderson, Northern States
R. Yang, EPRI
L. Liberatori, WOG
R. Montgomery, EPRI
R. Miller, W
C. Molnar, ABB - CE
W. Brunson, Frametome
S. Ray, W
R. McCardell, INEL
R. Hobbins, RRH Consulting
D. Diamond, BNL
A. Drake, WOG
M. Eyre, PECO
C. Beyer, PNNL
J. Mihalcik, BGE
G. Meyer, FCF
A. Azumi, Kansai EPCO
D. Risher, W
M. Gamolinski, IPSN
H. Blisr, ANL
B. Hopkins, Bechtel
A. Motta, Penn. State
D. Koss, Penn State
T. Link, Penn State

Introductory Remarks by the Subcommittee Chairman.

Dr. Powers convened the meeting at 8:30 am. He stated that the objective of the Subcommittee meeting is to gather information for the Advisory Committee on the use of very high burnup fuels. The Chairman noted that the Subcommittee is interested to understand how the foreign experiments performed in France, the former Soviet Union, and Japan are being followed up by the NRC staff, and the impact of the recent experimental results for reactivity initiated accidents on the NRC criteria for high burnup fuels in nuclear reactors.

Dr. Meyer, RES, noted that by the early 1990's, it had become clear that burnups in commercial power reactors were exceeding the burnup range for the validations of NRC's fuel behavior computer codes and related fuel damage criteria. Fuel suppliers were providing high burnup performance data to support the licensing of higher burnup fuel designs, but the NRC's independent capability had not been updated.

The present NRC licensing criteria for fuel behavior during reactivity transients involve two enthalpy values - one at 280 cal/g and the other at 170 cal/g. The 280 cal/g value is used as a limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired. The 170 cal/g value is used as an indicator of cladding failure for BWR reactivity transients initiated at zero or low powers; thermal margin criteria are used for all other BWR and PWR reactivity transients. Recently, cladding failure was observed at enthalpy values significantly below 170 cal/g at very high burnups.

The following are high burnup reactivity initiated accidents (RIAs) test results with cladding failure at low energy.

CABRI REP - Na - 1	62 GWD/T	failed at 30 cal/g
NSRR HBO - 1	50 GWD/T	failed at 60 cal/g
SPERT CDC-859	32 GWD/T	failed at 85 cal/g
NSRR HBO - 5	44 GWD/T	failed at 70 cal/g

The failure above was due to mechanical interactions unrelated to critical heat flux. There is a brief update of the results [NSRR (Japan)], for HBO - 5 (January 31, 1996) failure at approximate 70 cal/g deposited energy that resulted in vertical cracking over entire active fuel length, with most UO_2 pellets remained inside cladding. HBO-6 survived with approximate 100 cal/g energy deposition. For the CABRI (France) recent progress, REP - Na6 (March 1, 1996), survived with approximate 140 cal/g peak fuel enthalpy. REP - Na8 is scheduled for late summer 1996.

Dr. Meyer summarized some of the observations that confound the previous picture as follows:

- None of the failures in irradiated fuel occurred by DNB - related mechanisms except in the Russian Zr - 1%Nb cladding.
- HBO - 1, which failed at a low energy, exhibited significant ductility.
- Peak fuel enthalpies above ~100 cal/g do not appear possible for design - basis reactivity - initiated accidents.
- Power pulse widths in power rectors would be around 30-75 MSEC (full width at half the max. Power) whereas most RIA tests have been done with much narrower pulses.
- A fast (narrow) pulse causes the applied stress to be at its maximum when the cladding temperature is low.
- Fracture toughness for irradiated Zircaloy may be lower for cold zero - power accidents than for hot zero - power accidents.

- Zircaloy exposed for a long time to the reactor environment became brittle and this adversely affects its fracture behavior.
- UO_2 fragmentation and dispersal into the coolant are expected if the cladding fails above ~50 GWD/T.
- Consequences of fuel dispersal may be smaller because a limited amount of fuel is affected in these localized transients and the temperature of the particles is low (<<267 cal/g, UO_2 melting point).
- Departure from nucleate boiling is showing up in transient calculations at deposition energies below 100 cal/g.
- Transient fission gas release is high at burnups above ~50 GWD/t.

The implications of results to date are:

- Although cladding failure under RIA conditions may depend on a number of variables, failures at low fuel enthalpies cannot be ruled out based on the present data.
- Key variables appear to be cladding oxidation (and related hydriding), burnup level, cladding temperature, cladding material, and pulse characteristics.
- For broad power pulses or different materials, the failure mode might shift from a mechanical interaction to DNB - related phenomena.
- Fuel dispersal is likely to occur from failed fuel above ~50 MWd/T.
- Transient fission gas release up to 25% has been observed in recent tests and exceeds the 10% release assumed in Regulatory Guide 1.77.
- High energy fuel behavior involving molten fuel (i.e., >267 cal/g) should not be a concern for RIA's in power reactors because of low energy availability (<100 cal/g).
- Significant new data to refine the current picture will require several years to obtain.

Mr. Meyer briefly summarized the LOCA fuel/licensing criteria as : 2200 °F peak cladding temperature and 17% maximum cladding oxidation. Amount of oxygen equivalent to converting 17% of wall thickness to stoichiometric ZrO_2 (~100 microns). For cladding failure, 100% is assumed, Reg. Guides 1.3 and 1.4 specify certain percentages of the core inventory (TID source term). Some of the high burnups issues for LOCA are cladding embrittlement, stored energy, oxidation kinetics, swelling and rupture, fuel fragmentation, and source term.

Global Assessment of RIA Data - Mr R.K. McCardell

Mr R.K. McCardell, INEL, noted that there are two general phenomena that cause failure during simulated RIAs. These are cladding and fuel heat up; and expansion of fuel against cladding. The global assessment of RIA data is as follows:

- None of the failures in irradiated fuel occurred by DNB - related mechanisms except in the Russian Zr - 1% NB cladding.
- UO_2 fragmentation and dispersal into the coolant are expected if the cladding fails above approximately 50 GWD/T.

- Consequences of fuel dispersal (pressure pulses) may be small because the temperature of the particles is low (<<267 cal/g, UO_2 melting point).
- Transient fission gas release is high at burnups above 50 GWD/T.
- A fast (narrow) pulse causes the applied stress to be at its maximum when the cladding temperature is low.
- Departure from nucleate boiling is showing up in transient calculations at depositions energies below 100 cal/g.
- Fuel with burnup of 50 GWD/T has large radial power factor at rim of fuel pellet.
- Significant pressure pulses do not occur below 300 cal/g total energy depositions.

The CABRI Rep - Na Test Program - Dr F. Schmitz

Dr. Schmitz, Commissariat a l'Energie Atomique IPSN - France, summarized the principal findings and conclusions from the first five tests of the CABRI/RIA program. The decision to perform PWR/RIA tests in the CABRI facility and in particular in the sodium coolant environment of the CABRI test loop was based on the following:

- The French safety authority has asked in 1991 for experimental proof for the safe and satisfactory behavior of high burnup fuel under design basis accident conditions before authorizing the requested burnup increase from 47 to 52 GWD/T.
- The initial options to rely upon Japanese RIA tests in NSRR appeared to be unsatisfactory in 1992 because of the limited energy depositions capability and for unrepresentative fuel.
- The Japanese tests JM4 and JM5 (performed in 90 and 91) indicated PCMI failure occurring in the early phase of RIA when the cladding is still cold (20°C in NSRR).
- Limited instrumentation and diagnostics in NSRR lead to important uncertainties and resulting margins for use in safety demonstration.

With high burnup UO_2 fuel rods, REP - Na1 to 5, the major events that were identified are fuel rod failure, clad straining of unfailed rods, fission gas release, fuel fragmentation, fuel structure changes, and transient oxide spalling. The CABRI/RIA rests have provided a large amount of data for code validations and development. Dr. Schmitz stated that the CABRI/RIA, REP - Na experiments are fulfilling the initial objectives, namely;

- quantification of high burnup thermo - mechanical effects,
- evidence for increase PCMI loading,
- identification of new experimental findings beyond the previous data base.

Cladding Metallurgy and Ductility at High Burnup - Mr. H.M. Chung

Mr. Chung, Argonne National Laboratory, summarized the important aspects of cladding metallurgy at high burnup that influence fuel failure under RIA situations. The embrittlement mechanism of high burnup cladding are neutron displacement damage, hydriding, increased

oxygen in solution in alpha - phase, and irradiation. Higher burnup and hydriding decrease fracture toughness and increase ductile - brittle transition temperature (DBTT). High burnup cladding failure can be sensitive to pulse rate and initial pulse temperature, depending on DBTT. This phenomenon, unique to high burnup fuel, is insignificant in fresh or low burnup fuel because of low DBTT. Mr. Chung noted that high burnup fuel cladding exhibits unique mechanical properties that are sensitive to microstructure and temperature. Because of this, a failure correlation based on cladding properties will be far more accurate than the conventional correlations based on burnup. Initial database were obtained to correlate failure with several types of strain (ductility), however, understanding of associated microstructure is insufficient. Preventing or minimizing breakaway oxidations in cladding is probably the most practical and the best way to improve the performance of high burnup fuel.

Power Excursion Analysis for LWRs - Dr. David Diamond

Dr. Diamond, Brookhaven National Laboratory, summarized the results of BWR rod drop accident (RDA) and PWR rod ejections accident (REA) calculations.

The RDA calculations assumed initial thermal - hydraulic conditions and control rod pattern so that the worth of a control rod dropping out of the core was approximately 950pcm. The maximum increase in fuel bundle enthalpy in the core was found to be less than 70 cal/g for the medium burnup core and less than 100 cal/g for the high burnup core. This is low relative to existing acceptance criteria for this event. However, it is larger than what might be of interest in high burnup fuel as a result of the recent experimental tests - namely, 30 gal/g. The enthalpy rise was determined not only by the dropped rod worth and the magnitude of the feedback but also by the timing of the feedback. With large subcooling, the generation of void feedback is delayed, and the fuel enthalpy continues to rise after the initial increase in enthalpy due to the power pulse.

The calculations were used to determine the increase in fuel enthaply as a function of burnup and the effect of certain modeling assumptions. The results of the calculations were consistent with the expectation that the peak fuel enthalpy in any axial location of any bundle would be a complicated function of the control rod worth, the distance of the bundle from the dropped control rod, and the burnup at that location. The implication of this is that high burnup in a fuel bundle does not inherently limit the enthalpy rise.

The uncertainties in modeling that are particular importance with high burnup fuel are the result of the rim effect. The rim effect is the large increase in plutonium concentration and power along the surface of the fuel pellet.

For the REA in a PWR, the rod insertion limits determines the maximum rod worth. Rod worth are less than 1%. The power pulse width are approximately 30 - 75 m. seconds.

Failure of Zircaloy Cladding Under Severe Loading Conditions - Mr. Todd Link, Mr. Donald Koss, Mr. Arthur Motta

Mr. Todd Link, Pennsylvania State University, stated that studies are underway at Penn. State to understand Zircaloy fracture behavior under severe loading conditions. The potential failure modes of cladding are failure by crack initiations and growth, general damage and plasticity, and deformation localization (necking instability).

The model of cladding failure by necking instability is based on the formations of a small non - uniformity (defect) which limits the ductility of the cladding. The imperfection acts to trigger instability. The ductility of thin - walled irradiated cladding is very sensitive to the presence of small imperfections in thickness such as from a cracked hydride or oxide layer. Zircaloy exposed for a long time to reactor environment may develop a brittle rim (or thickness variation) which can act as an imperfection, severely limiting cladding ductility in an RIA event.

NEI Presentation - Mr. George Wu

Mr. Wu summarized NEI's position and evaluation of reactivity insertions accidents involving high burnup fuel. In 1994, information was distributed by the NRC to the nuclear industry on preliminary results from foreign research on high burnup fuel integrity during RIAs. The NRC staff requested information from the industry to assist in assessing the safety significance of fuel failure in the tests. In response, NEI submitted an industry safety assessment in December 1994, together with data on fuel burnup levels at several licensee facilities. Additional fuel burnup data was submitted in February 1995.

The industry assessment had shown that RIAs are highly unlikely and that the radiological consequences from postulated high burnup fuel failure in RIAs are limited. NEI, however, stated in the assessment that since the test results are preliminary and their applicability to U.S. light water reactor fuels remain to be confirmed, a more detailed evaluation of the data is necessary in order to achieve a final resolution of the staff concerns.

More recently, additional tests have been conducted in the foreign research programs. This more recent information has been assessed by the industry to further evaluate the relevance of various test conditions, and of the earlier failures of fuel segments in the tests, to LWR conditions. Based on this evaluation, NEI does not believe that any revision of the existing RIA fuel failure criteria by the NRC is justified.

A detailed evaluation of the data form RIA simulation experiments has been conducted by EPRI and made available to NEI. This evaluation showed that the test conditions are conservative compared to U.S. LWR, and that the failure data are not directly applicable to commercial LWR fuels. Further, the evaluations showed that RIAs are not a limiting constraint in high burnup fuel behavior. NEI issued a position paper and submitted it to the NRC staff on April 8, 1996 regarding this issue.

EPRI Presentation - Mr. Odelli Ozer

Mr. Ozer summarized the evaluation of data from RIA simulation experiments. The general approach includes evaluation of experimental data and analysis that involves calculation of expected fuel responses under test conditions. The analytic approach of an anticipated in-reactor conditions resulting from a hypothetical RIA event is a coupled 3-D neutron kinetics/thermal-hydraulic (CORETRAN) calculations. The general observations are predictions with 3-D models are significantly less severe than 1-D (FSAR) models and the pulse magnitudes are consistent with vendor 3-D analyses. The RIA tests included in the evaluation are SPERT-CDC (U.S.), NSRR-BWR & PWR (Japan), and CABRI-REP Na (France).

The general observations are the failure mechanism for low-exposure fuel is thermally driven; and for irradiated fuel, possible failure mechanism is driven by PCMI. The RIA test conditions are more severe than LWRs conditions. Failures in RIA tests have been caused by unrepresentative experimental conditions and/or faulty test specimens. The region defined by successful tests can be used estimating the response of irradiated fuel to a RIA event. No need is evident for modification of existing failure criteria. If fuel failures result in dispersal of pellet fragments into the coolant, the pressure increase is benign.

NSRR Experiments with High Burnup Fuel - Dr. Toshio Fujishiro (Japan)

Dr. Fujishiro, Japan Atomic Energy Research Institute, stated that behavior of reactor fuels during off-normal and postulated accident conditions such as RIA is being studied in the NSRR program of the Japan Atomic Energy Research Institute (JAERI) to provide a data base for the regulatory guide of LWRs. Numerous experiments using pulse irradiations capability of the NSRR have been performed to evaluate the thresholds, modes, and consequences of fuel rod failure in terms of the fuel enthalpy, the coolant conditions, and the fuel design.

The HBO-1 test with a 50 MWd/KgU PWR fuel resulted in fuel failure at the energy deposition of approximately 60 Cal/g fuel. The results suggested possible reduction of failure threshold for high burnup fuels, and indicate that PCMI with swelling of the fuel pellets leads to the failure. Rapid thermal expansion of accumulated fission gas can intensify the swelling and fissions gas release, and subsequent fuel fragmentation.

The 50 Mwd/KgU fuel rods in three experiments following the HBO-1 test survived through the transients with peak fuel enthalpy ranged from 37 to 74 Cal/g fuel. However, significant fission gas release up to 22.7% occurred.

Further investigations on fuel failure mechanisms through in-pile integrated experiments, out-of-pile separate effect tests and phenomenological modeling could contribute to accident-conscious fuel design to avoid fuel failure and excessive fission gas release.

Westinghouse Owners Group - Mr. Lewis F. Liberatori, Jr.

Mr. Liberatori, Jr., Consolidated Edison Co. N.Y., Inc., briefly described the high burnup fuel RIA assessment for PWR. He noted that typical PWR analysis methods include conservative assumptions such as:

- Reactor is just critical
- Control rods are inserted to insertion limit
- Ejected rod is one of the inserted rods
- Adverse Xenon or burnup distribution to maximize rod worth
- Conservative hot rod heat transfer, reactor trip point, and trip reactivity
- Conservatively assume constant post-ejection power peaking factor

The 3-D kinetics codes are available, but not typically used or licensed. The PWR analysis calculations are used to determine limiting plants/cycles, and to obtain peak Cal/gm versus burnup. The identified limiting cases are composite 3-loop plant with very high Δk in a medium burnup 18-month cycle, and a composite 4- loop plant with high Δk in a proposed high burnup 24-month cycle.

The 3-D analytical methods show that the peak fuel enthalpy is not likely to exceed 60 Cal/gm at any burnup. Results showed only about 2-7% of fuel rods have both a burnup exceeding 30-40 GWd/MtU and a peak fuel enthalpy exceeding 30 Cal/gm. Results are consistent with those presented by others at the Caderache, France, which showed a factor of 2-4 benefit for 3-D analysis. With the use of 3-D analysis, the DNB criterion is expected to be limiting with respect to assumed cladding failures. No need to revise fuel failure limit. More realistic dose evaluation methods show that plants can assume 100% fuel cladding damage without exceeding licensing limits.

PECO Energy - Mr. Matt Eyre

Mr. Eyre described the BWR safety assessment for RIA and high burnup fuel. He described the control rod drop accident (CRDA) with the conclusion that it is a very low probability event (less than 1×10^{-12} /yr), with an energy deposition of less than 15 Cal/gm at high burnup (> 60 GWD/Mt). The dose consequences are well within 10 CFR Part 100 limits. The CRDA power pulse is smaller and slower than simulated test pulses.

For transients, the fuel response is fundamentally different from CRDA with the pulses smaller and slower. The expected power oscillations resulting form thermal-hydraulic instabilities could be mitigated by stability solutions designed to protect MCPR safety limit, with energy deposition less than 1 Cal/gm. The power oscillation events with high burnup fuel does not introduce new BWR safety concerns.

Control Rod Insertion Problems - Ms. Margaret S. Chatterton (NRC)

Ms. Chatterton stated that recently, the NRC has issued an Information Notice (96-12) and a Bulletin (96-01) to alert the licensees to problems encountered during recent events in which control rods failed to completely insert upon the scram signal, and to assess the operability of control rods particularly in high burnup fuel assemblies. Ms. Chatterton mentioned three specific events:

- The first occurred on December 18, 1995, with South Texas Unit 1 at 100% power, a pilot wire monitoring relay actuation caused a main transformer lock out which resulted in a turbine trip and reactor trip. While verifying that control rods had inserted fully after the trip; operators noted that the rod bottom lights of three control rod assemblies were not lit. The digital rod position indication for each rod indicated six steps withdrawn. During subsequent testing of all control rods in the affected banks, the rod position indication for the same three as well as a new location indicated six steps withdrawn. Within an hour after the tests, two of the rods drifted to rod bottom positions and the other two were manually inserted. All four control rods were located in fuel assemblies that were in their third cycle with burnup greater than 42,800 Mwd/MTU.

- The second event occurred on January 30, 1996 at Wolf Creek Plant. After a manual scram from 80% power, five control rod assemblies failed to insert fully. Three of the affected road drifted to fully inserted within 20 minutes, one within 60 minutes and the last within 78 minutes. The five rods were all in 17 X 17 vantage fuel with burnup greater than 47,000 MWd/MTU.

- The third event occurred on February 21, 1996 at the North Anna plant. During the insert shuffle in preparation for loading North Anna 1, cycle 12, two new control rod assemblies could not be removed with normal operation of the handling tool from the fuel assemblies in the spent fuel pool in which they were temporarily stored. The two affected fuel assemblies were vantage fuel with burnups of 47,782 and 49,613 MWd/MTU.

Ms. Chatterton noted that the NRC staff is concerned regarding the above events that represent potential precursors and apparent correlation of problems with high burnups and drag forces, rod-drop time histories, and lack of rod recoil. The staff is interacting with PWR Owners Group to determine the root cause of these events and continues to assess industry responses to the NRC Bulletin. The staff will issue generic communications as required.

Future Action

As a result of the Subcommittee discussion, the Subcommittee Chairman raised the following concerns and issues,

- NRC does not plan to undertake its own experiments, nor does its involvement in foreign experimental efforts extend to definition or control of the experiments.
  The U.S. industry plans no testing.

- High burnup fuel does not comply with fuel limits specified in the Standard Review Plan.

- The computational tools available to the NRC for analyses (such as the FRAPCON and FRAP-T codes) have not been updated.

- Revised severe accident source term (NUREG-1465) explicitly excludes high burnup fuel and there is no effort to obtain data to amend this source term so it can be used to treat high burnup fuels.

- The immediate threats to public health and safety due to the susceptibility of high burnup fuel to degradation during the transient overpower events and effects that arise in other types of off normal events should be examined further.

The Reactor Fuels Subcommittee intends to follow-up on the high burnup issues closely.

Background Material Provided to the Subcommittee Prior to this Meeting.

1. NRC Information Notice 94-64.
2. NRC Information Notice 96-12.
3. NRC Bulletin 96-01.

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20006, (202) 634-3274, or can be purchased from Neal R. Gross & Co., Inc., Court reports and Transcribers, 1323 Rhode Island Avenue, N.W., Washington, D.C. 20005, (202) 234-4433.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
5) LCO 3.4.5.a, Control Room Air Treatment System Includes: LCO 3.6.2.a.(12), Table 3.6.21, Control Room Air Treatment System Initiation. b. When conducting scram time testing in conjunction with system leakage or hydrostatic testing, specification 3.7.1, Special Test Exception - Shutdown Margin Demonstrations, shall be met. c. The drywell and the pressure suppression chamber are intact with at least one door in each personnel air-lock closed. d. With the requirements of specification 3.7.2.a, 3.7.2.b, or 3.7.2.c not satisfied, immediately abort system leakage or hydrostatic testing and scram time testing activities and reduce the average reactor coolant temperature to $\leq 212°F$ within 10 hours.	

BASES FOR 3.7.2 AND 4.7.2 - SYSTEM LEAKAGE AND HYDROSTATIC TESTING

This special test exception allows the reactor coolant temperature to be considered less than 212°F (i.e., cold shutdown) when the reactor coolant temperature is greater than 212°F (i.e., hot shutdown) but less than 275°F while performing reactor vessel system leakage or hydrostatic testing and scram time testing. This allows operational flexibility since temperatures may exceed 212°F during the test and can drift higher since decay and mechanical heat do not allow for exact control. Not all Limiting Conditions of Operation (LCOs) applicable to operation at coolant temperatures > 212°F, e.g. Primary Containment and Containment Spray, apply during this Special Test Exception. Additionally, because reactor vessel fluence increases over time, this testing will require greater coolant temperatures. The requirement for reactor building integrity, the conditions placed on the primary containment, and the cold shutdown requirements for other plant systems provides conservatism in the response of the unit to an operational event. Shutdown margins need only be demonstrated when performing scram time testing in conjunction with system leakage or hydrostatic testing (i.e., shutdown margins need not be demonstrated when performing a pressure test only). This special test exception is to be used when low decay heat values are present (e.g., such as following an outage) and reactor coolant activity levels are within the limits specified in Specification 3.2.4.

ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. DPR-63
DOCKET NO. 50-220

Marked Up Copy of Technical Specifications

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

LIMITING CONDITION FOR OPERATION

d. If Specifications a, b and c are not met, a normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.

e. During reactor operation, except during core spray system surveillance testing, core spray isolation valves 40-02 and 40-12 shall be in the open position and the associated valve motor starter circuit breakers for these valves shall be locked in the off position. In addition, redundant valve position indication shall be available in the control room.

f. Whenever irradiated fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F, two core spray subsystems shall be operable except as specified in g and h below. If in Special Test Exception 3.7.2, one subsystem shall be operable in each system.

g. If one of the above required subsystems becomes inoperable, restore at least two subsystems to an operable status within 4 hours or suspend all operations that have a potential for draining the reactor vessel. In addition, if in Special Test Exception 3.7.2, immediately abort system leakage or hydrostatic testing and scram time testing activities and reduce average reactor coolant temperature to ≤ 212°F within 10 hours.

SURVEILLANCE REQUIREMENT

d. Core spray header ΔP instrumentation

check	Once/day
calibrate	Once/3 months
test	Once/3 months

e. <u>Surveillance with Inoperable Components</u>

When a component becomes inoperable its redundant component or system shall be verified to be operable immediately and daily thereafter.

f. With a core spray subsystem suction from the CST, CST level shall be checked once per day.

g. At least once per month when the reactor coolant temperature is greater than 212°F, verify that the piping system between valves 40-03, 13 and 40-01, 09, 10, 11 is filled with water.

BASES FOR 3.1.4 AND 4.1.4 CORE SPRAY SYSTEM

Based on the limited time involved in performance of the concurrent refueling maintenance tasks, procedural controls to minimize the potential and duration of leakage and available coolant makeup (CST) provides adequate protection against drainage of the vessel while the suppression chamber is drained.

Specification 3.1.4e establishes provisions to eliminate a potential single failure mode of core spray isolation valves 40-02 and 40-12. These provisions are necessary to ensure that the core spray system safety function is single failure proof. During system testing, when the isolation valve(s) are required to be in the closed condition, automatic opening signals to the valve(s) are operable if the core spray system safety function is required.

In the cold shutdown and refuel conditions, the potential for a LOCA due to a line break is much less than during operation. In addition, the potential consequences of the LOCA on the fuel and containment is less due to the lower reactor coolant temperature and pressures. Therefore, one subsystem of a core spray system is sufficient to provide adequate cooling for the fuel during the cold shutdown or refueling conditions. Therefore, requiring two core spray subsystems to be operable in the cold shutdown and refuel conditions provides sufficient redundancy. One subsystem in each system is required to be operable to provide sparger redundancy while in Special Test Exception 3.7.2.