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GRIESLER,C.W. Wisconsin Public Service Corp.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Requests exemption from 10CFR50,46 re proposed Amend 64 to Tech Specs,Proprietary "Evaluation of Margin Between App K Evaluation Model & Best Calculations of Large Break LOCA for Kewaunee Nuclear Unit," encl,Rept withheld,Fee paid.

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NRC-85-62

WISCONSIN PUBLIC SERVICE CORPORATION
P.O. Box 1200, Green Bay, WI 54305

March 29, 1985

Dr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Request for Exemption from 10 CFR 50.46

This letter and the attachments hereto provide the basis for and request by Wisconsin Public Service Corporation (WPSC) for an exemption for the Kewaunee Nuclear Power Plant (KNPP) from 10 CFR 50.46(a)(1) and item I.A of Appendix K to 10 CFR 50. We initially informed you of our intent to request this exemption on March 18, 1985.[1] The exemption is requested while technical work proceeds to resolve concerns raised by the staff during its review of Proposed Amendment 64 to the KNPP Technical Specifications, and while KNPP resumes normal, full power operation. The staff's concern is that the current evaluation models used by Exxon Nuclear Company (ENC) are not based on adequate sensitivity studies, and consequently may not demonstrate compliance with Emergency Core Cooling System (ECCS) criteria. The technical work to revise those models, when completed, will demonstrate that KNPP is in full compliance with all ECCS criteria.

WPSC submitted Proposed Amendment 64 on November 30, 1984.[2] The proposed changes affected the nuclear heat flux peaking factors specified in Section 3.10.b of the KNPP Technical Specifications. The changes were based on calculations performed by WPSC's fuel vendor, ENC, in accordance with evaluation models approved or under review by the staff. Several questions were raised by the staff[3] during its review of Proposed Amendment 64. WPSC has resolved all but one of these questions.[4]

In addition, on March 20, 1985, the staff informed WPSC by telephone of several concerns in regards to Exxon Nuclear Company's Evaluation Models and ENC's controls for these models. Although not directly related to the review of Proposed Amendment 64, the staff notified WPSC because of the potential impact these concerns could have on our proposed amendment. The concerns discussed

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were the use of heat transfer augmentation factors, a coding error associated with these factors, and an input error on an Evaluation Model calculation performed for another licensee.

To address these additional concerns, WPSC immediately performed an audit of ENC's ECCS methodology as related to KNPP's analysis. The results of this audit were submitted to the staff on March 26, 1985.[5] In summary, the audit concluded that:

1. The heat transfer augmentation factors were used in the initial calculations performed in support of the KNPP proposed amendment. These calculations were performed in early 1984, and the use of heat transfer augmentation factors was appropriate at the time. These factors were not used in the reanalysis performed in February, 1985 in response to the staff's request for additional information.[3,4]

2. The computer code error did occur on the initial KNPP analysis, but it had only a minor effect on the analysis. The error was evaluated and found to be worth approximately 50°F in Peak Clad Temperature (PCT) when augmentation factors are not used. When heat transfer augmentation factors are used, the error was determined to be less than 1% of the total augmentation and worth less than 10°F in PCT. The coding error also was corrected in the reanalysis.

3. There is reasonable assurance that the input data used in the KNPP calculations are correct. The specific input error which occurred on the other licensee's docket has been checked for KNPP; it did not occur in the KNPP analysis.

In addition to the review of the specific concerns raised by the staff, the audit team also reviewed the controls ENC provides for Evaluation Models. The team concluded that significant improvements have been made in these controls as a result of and in response to NRC inspections which occurred in April and September of 1984. Consequently, it can be concluded that the single remaining issue associated with Proposed Amendment 64 is the K(Z) function (figure TS 3.10-2).

K(Z) is a function employed to define the maximum power peaking factor as a function of elevation in the core. Administratively limiting the peaking factor to the value specified in the technical specifications through K(Z) as determined by Evaluation Models shows that the peak clad temperature (PCT) will not exceed 2200F, as required by 10 CFR 50.46.

Exxon Nuclear Company developed an ECCS Evaluation Model (WREM PWR) applicable to pressurized water reactors (PWR) in 1975. Existing models and codes, largely from Atomic Energy Commission activities, were used for this initial development. During the past decade the WREM PWR ECCS Evaluation Model has been modified to handle a large range of PWR reactor types, to utilize improved and more efficient calculational techniques, and to incorporate new data relating to ECCS phenomena as these data became available. It was recognized in the beginning that results of ECCS analysis for bottom reflood plants are sensitive

to axial power peaks occurring near the top of the core due to the poorer reflood cooling in this region. The effect was addressed by Westinghouse in the form of the K(Z) function on the allowed Fq operating limit. Since the bottom reflood behavior is primarily a system effect, the NRC staff had accepted the argument until recently that the required axial power dependence should be relatively independent of differences between the ENC fuel and the Westinghouse fuel designs for specific reactors. Thus, the Westinghouse developed K(Z) function was accepted for Exxon Nuclear fuel as well as Westinghouse fuel without additional sensitivity studies. ENC ongoing model development activities concentrated on the center peaked cosine axial power shape resulting in excessive conservatisms in the upper region of the core.

At the NRC staff's request, activity was initiated by Exxon Nuclear in mid-1984 to determine the axial power shape dependence for a Westinghouse 3-loop PWR plant. It was found that to assure compliance with the ECCS criteria using the approved ECCS Evaluation Model, the dependence of the maximum linear heat generation rate (LHGR) on core height needed to be revised. A calculated ENC K(Z) function has been inserted in the Technical Specifications for this plant. In January 1985, it was found that the LHGR dependence on core height required a larger reduction in LHGR limits for a Westinghouse PWR 4-loop plant with ice condenser containment, than was required for the 3-loop plant discussed above. Although Technical Specification changes have not yet been proposed for this plant, the licensee has implemented administrative controls for operation in conformance with a more restrictive limit. A similar calculation for the Kewaunee Nuclear Power Plant indicates that power reductions of approximately 15% would be required to maintain PCT in acceptable limits using the currently approved ENC Evaluation Model.

As explained below, WPSC believes that a sufficient technical basis exists to allow NRC to conclude that an exemption from 10 CFR 50.46 is justified which would allow operation of the KNPP under the existing K(Z) function. The requested exemption is proper under 10 CFR 50.12(a) in that all the evidence shows granting it will not endanger life, public health and safety, the environment, or property, and the exemption is otherwise in the public interest. This conclusion is based on test data from experiments and best estimate calculations performed in the past several years which indicate the extremely conservative nature of evaluation models, and the extremely low probability that a large break LOCA would occur. In addition, the exemption is needed to allow KNPP to operate at full power for the benefit of its customers, and to provide for the efficient use of the already strained resources of the staff, WPSC and ENC.

There is currently general agreement in the technical community that the PCT's calculated by Appendix K evaluation models overestimate actual PCT's by 1000 to 1200 degrees F. This agreement is based on, among other things, the conservatism known to exist in ECCS Evaluation Models, data obtained from experiments and so-called "best-estimate' calculations.

ENC has long recognized that significant conservatism exists in the bounding reflood heat transfer correlations which were required to obtain approval of the 1975-1976 ECCS evaluation models. The EXEM/PWR ECCS model submitted for NRC review in 1982 contains revised reflood heat transfer correlations for both 15 x 15 and 17 x 17 PWR fuel assemblies. This model is currently

under active NRC staff review and alterations are being made to the heat transfer correlations in order to obtain NRC approval. Exxon Nuclear is just finishing recorrelation of reflood heat transfer coefficients based on FLECHT data for the 4-loop Westinghouse PWR plant discussed previously and has identified significant heat transfer improvement at the upper core elevations. Similar conservatisms are expected to be evident when an analysis is performed for the KNPP using more appropriate reflood heat transfer correlations.

A major additional conservatism should also be recognized. The 2-loop Westinghouse reactors, such as KNPP, utilize ECCS injection directly into the reactor vessel upper plenum. Both test data and best estimate calculations for this type of reactor show core cooling occurring from a combination of bottom reflood and top down quenching. The test data and the calculations also show substantial margins to the ECCS acceptance criterion for this type of reactor. Requirements for implementing new ECCS analysis models which specifically consider upper plenum injection have been issued by the NRC and plans and schedules to implement this requirement are being formulated. While the overall effects of upper plenum injection (UPI) on peak clad temperatures have not yet been quantified, it is clear that the top region of the core will experience much better cooling with upper plenum injection than with only bottom reflood. Thus, when an appropriate UPI analysis is performed the allowed limits in the top of the core will be much less restrictive than the current evaluation models predict.

The conservative nature of Evaluation Models is evident from best-estimate studies performed by many parties. For example, Combustion Engineering has performed analyses which indicate that the actual peak clad temperature for a typical PWR is predicted to range from 1000°F to 1200°F.[6] Similarly, Westinghouse Electric Corporation has informed WPSC that their best estimate calculations, and best estimate calculations performed by many others yield similar results: actual peak clad temperatures are predicted to range from several hundred to more than 1000°F below the acceptance criterion.[7]

These efforts have also shown that these results apply for 2 loop plants, similar to the Kewaunee Nuclear Power Plant. Sandia National Laboratory performed best estimate LOCA calculations for a Westinghouse 2 loop plant. As noted by Bajorek and McIntyre,

> The study conducted at Sandia National Laboratory investigated the behavior of coolant injected into the upper plenum of the two loop vessel. Although the intent of the study was not to address Appendix K conservatisms, the results clearly indicated significant margin to the 2200 degree F Appendix K limit.[7]

Bajorek and McIntyre also address upper plenum injection concerns and the applicability of test data recently obtained from the CCTF and SCTF facilities in Japan. They conclude that the test results do not warrant an additional penalty, and indicate that the effects of upper plenum injection may lead to a benefit in calculated PCT.[7]

Finally, Westinghouse has performed extensive sensitivity studies to investigate the effects of various analytical models on LOCA analysis results and to quan-

tify the LOCA margin available in a two-loop plant. They concluded that even retaining many of the conservatisms of an Appendix K evaluation model in a "better estimate" calculation, there is 286°F margin to the 2200°F limit.[7] This analysis is applicable to KNPP, and shows that with a more appropriate model, KNPP would be in compliance with ECCS criteria.

Based on this discussion, there is sufficient evidence to conclude that the actual peak clad temperature expected to occur at KNPP following a hypothetical loss of coolant accident would be several hundred to 1000 degrees below the regulatory limit of 2200°F. The problem at hand (that is, the current ENC evaluation model concerns) is a result of the conservatisms in the analysis: it is an artifact of the method employed.

While this information alone provides a sufficient technical basis for approval of WPSC's exemption request, it is equally important to note that these analyses are of events POSTULATED to occur. The probability of these events, (namely, a loss of coolant accident) and the probability of severe consequences due to these events, have been predicted to be vanishingly low.

For example, the probability of a double ended guillotine break of reactor coolant loop piping has been estimated to be from 10^{-5} breaks per reactor year down to 7×10^{-12} breaks in a reactor lifetime.[8] In reviewing analyses submitted by a utility group in regards to generic issue A-2, "Assymetric Blowdown Loads on PWR Primary Systems", the staff concluded

> ...that large margins against unstable crack extension exist for stainless steel PWR primary main loop piping postulated to have large flaws and subjected to postulated SSE and other plant loadings.[9]

In other words, the staff concluded that these piping systems would "leak before break", and that a large break need not be considered as a design basis to resolve this safety issue.

The low probability of a LOCA notwithstanding, additional studies have been performed to estimate the probability of significant consequences from a LOCA. The American Nuclear Society's Special Committee on Source Terms has judged that combining the probabilities of fuel damage, containment failure, weather conditions and the presence of a large number of people, the probability of significant consequences is on the order of 10^{-8} per reactor year. The committee's major findings include that

> estimates of source terms associated with severe core damage accidents can be reduced from estimates in WASH-1400 by more than an order of magnitude to several orders of magnitude.

The committee also found that:

> For large dry PWR containments, sufficient information exists to support the calculation of source terms ranging from a small fraction of a percent to no more than a few percent of the core inventory of important fission products species.[10]

Therefore, based on the low probability of a LOCA, and the subsequent low probability of release of substantial amounts of radioactivity in a manner that could affect the health and safety of the public, it can be concluded that the Kewaunee Nuclear Power Plant can be operated at full rated capacity with reasonable assurance that the health and safety of the public is maintained.

As shown above, granting the requested exemption will not endanger life, health, public safety, property or the environment. In addition, the public interest favors granting the exemption. First, there is the cost of not receiving an exemption. If WPSC does not receive the exemption, KNPP operation could be limited to about 85% power. Based on the estimated time to reanalyze (approximately 8 months--until December, 1985,--as indicated on the attached schedule), the cost of this derate is estimated to be $6.8 million. This estimate includes only the cost of replacement power, and not the costs of interim analyses and the expenditure of internal resources to revise and implement interim procedures. This cost will be borne by WPSC's rate-payers. Given that this is not considered to be a technical problem, this cost to consumers is unwarranted.

Second, granting the requested exemption would provide for the efficient use of the resources of all parties involved. Again, given the evidence to the contrary, it would be a severe and injustice to require a derate of KNPP. The facility is already built. It can be operated safely. It would be wastefully inefficient not to fully utilize it. Furthermore, a derate would force WPSC to perform interim analyses to accurately define the extent of the derate and gain as much margin as is immediately available. It would require ENC resources to perform the analyses and staff resources to review them. This process would divert resources from ENC's completion of the reanalyses of KNPP using a new ECCS model, which reanalysis is expected to show compliance at full power. If the exemption were granted, this process would be required only once, and would be completed in a more efficient manner.

Third, the derate might create an unwarranted perception on the part of the public about the safety of nuclear power--a perception that could have adverse effects on the nuclear industry and the nation's energy policy. An announcement that KNPP, a plant recognized for its safe and reliable operation by the NRC and industry alike, had to endure a derate because of "safety concerns" might help cause or contribute to many ill effects. For example, the financial health of WPSC and other investor-owned utilities with nuclear interests could be harmed as investors shunned them in the face of perceived safety concerns. The message that such a derate would convey to the financial community, other nuclear operators, and the public in general, when not technically justified, should not be sent. While WPSC has always placed safety in the forefront of its operations, a derate without technical basis does not further safety and is unwarranted.

Finally, in seeking this exemption WPSC is not asking permission to do anything different from what it has done in the past. We are requesting to maintain the status quo until more accurate computer models can be constructed. Granting the exemption will not produce any new discharges, will not have any environmental consequences (irreversible or otherwise), will not increase the probability of an accident of any type, and will not reduce present margins of safety. The

exemption will simply allow WPSC to continue to operate KNPP exactly as we have in the past, and that method of operation has proven safe, reliable and extremely cost effective for Wisconsin consumers.

It should be noted that in general, the staff has agreed with the technical basis of the requested exemption. In Secy-83-472, the staff notes the following:

> In order to justify continued operation of their plant in such situations, licensees have expended considerable engineering resources proposing and making modifications to their analysis models to offset the PCT increases and bring the calculated PCTs back below the 2200°F limit. This in turn consumes considerable staff resources during the review and approval process to assure compliance with the regulations. For the most part, these efforts have resulted in a negligible impact on actual plant safety, since the outcome of such exercises has usually not involved any significant change in operational flexibility or any operational limit.
>
> Recent analyses indicate that the most probable PCT that would be experienced during the limiting large LOCA would be 1000°F to 1200°F for both BWRs and PWRs. These results have been obtained from advanced computer codes (TRAC and RELAP5), developed independently at two separate national laboratories. Industry calculations with realistic LOCA computer codes reach the same conclusion; namely that there is approximately a 1000°F to 1200°F margin between the PCT expected during the limiting large break LOCA and the 10 CFR 50.46 limit of 2200°F. These analytical estimates are now well verified for both classes of reactors by the LOFT and TLTA experiments.
>
> Based on the above, the staff has concluded the following:
>
> 1. The safety margin in peak cladding temperature provided by current evaluation models to assure compliance with 10 CFR 50.46 limits is approximately 1000°F to 1200°F for the large break LOCA.
> 2. This margin is more than adequate to assure successful ECCS performance in the event of a LOCA.
> 3. This margin can be reduced without adverse effect on plant safety.
> 4. Acceptable reduction in this margin may be warranted to avoid unnecessary restrictions in operation as a result of excessive conservatism imposed in ECCS evaluations.[11]

In summary, WPSC believes that adequate information exists for the staff to conclude that there is reasonable assurance that KNPP can operate at full power within existing limits and controls as specified in the KNPP technical specifications without undue risk to the life, health and safety of the public, property or the environment. In addition, the public interest favors this request because the cost of not obtaining this exemption, including the cost of replacement power and the resources which would be expended by WPSC, ENC and the staff, would be substantial.

ENC has developed a schedule for performing the necessary analyses which is attached.[12] As currently envisioned, it is expected that this work would be completed by the end of December, 1985. WPSC believes that this schedule is appropriate, and allows for the ultimate resolution of this issue in an orderly fashion.

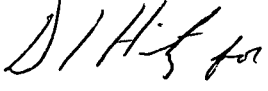
Therefore, WPSC respectfully requests that:

1. In accordance with 10 CFR 50.12(a), the NRC grant WPSC a temporary exemption to the requiremens of 10 CFR 50.46 as it relates to K(Z), and

2. The NRC approve Proposed Amendment 64 to the KNPP Technical Specifications, as submitted.

As the KNPP is scheduled to resume power operation on April 9, 1985, we request your prompt review of this item. In accordance with the requirements of 10 CFR 170, a check for $150.00 is attached as an application fee.

Very truly yours,



Carl W. Giesler
Vice President - Power Production

CAS/jks

cc - Mr. S. A. Varga, US NRC
Mr. Robert Nelson, US NRC
Mr. Trevor Will, F&L

Subscribed and Sworn to
Before Me This 29th Day
of March 1985

Notary Public, State of Wisconsin

My Commission Expires:
June 28, 1987

List of References

1. Letter from D. C. Hintz (WPSC) to S. A. Varga (US NRC) dated March 18, 1985, providing advance notification of WPSC's intent to request an exemption to 10 CFR 50.46.

2. Letter from C. W. Giesler (WPSC) to H. R. Denton (USNRC) dated November 30, 1984, submitting Proposed Amendment 64 to the KNPP Technical Specifications.

3. Letter from S. A. Varga (USNRC) to D. C. Hintz (WPSC) dated February 5, 1985, requesting additional information on Proposed Amendment 64 to the KNPP Technical Specifications.

4. Letter from D. C. Hintz (WPSC) to S. A. Varga (USNRC) dated February 26, 1985, providing additional information on Proposed Amendment 64 to the KNPP Technical Specifications.

5. Letter from D. C. Hintz (WPSC) to S. A. Varga (USNRC) dated March 26, 1985, providing a summary of an audit performed by WPSC of Exxon Nuclear, Inc.

6. Telecopy memo from David Dixon (Combustion Engineering) to D. J. Ropson (WPSC), providing information on C-E's best estimate calculations; attachment 1.

7. Letter from E. V. Somers (Westinghouse Electric Corporation, Inc.) to D. J. Ropson (WPSC), dated March 25, 1985, providing a paper by Bajorek and McIntyre entitled "An Evaluation of the Margin Between Appendix K Evaluation Model and Best Estimate Calculations of Large Break Loss-of-Coolant Accidents for the Kewaunee Nuclear Unit," attachment 2.

8. Regulatory Analysis of Mechanistic Fracture Evaluation of Reactor Coolant Piping, A-2 Westinghouse Owners Group Plants, NRC Generic Letter 84-04.

9. NRC Generic Letter 84-04, Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, February 1, 1984.

10. Report of Special Committee on Source Terms, American Nuclear Society, September, 1984.

11. Secy-83-472, Emergency Core Cooling Systems Analysis Methods, November 17, 1983.

12. Proposed Schedule for Kewaunee LOCA/ECCS Reanalysis (with Current Model Conservatisms Removed) to confirm K(Z), attachment 3.

Attachment 1 to WPSC's

Exemption Request

March 29, 1985

Reference 6

Telecopy Memo from Dave Dixon, Combustion Engineering

to Dan Ropson, Wisconsin Public Service Corporation

March 22, 1985

Telecopy To: Dan Ropson, WPSC

From: David Dixon; Combustion Engineering, Inc.

The results of large break LOCA evaluation model (EM) analyses generally show a predicted peak cladding temperature (PCT) sensitivity to axial power shape. This sensitivity is relatively small compared to the PCT margin produced by the Appendix K conservatisms in current NRC approved EMs for PWRs.

The C-E EM produces a variation in PCT as the axial power shape is varied. A preliminary calculation for a generic W four-loop plant demonstrates that a cosine power distribution peaked at the core mid-plane (6 foot elevation) produces a slightly higher PCT than a power distribution with the same Fq.(2.4) skewed upwards to about the 8 foot elevation. The study produced a PCT difference of less than 50°F for both of the possible reflood heat transfer conditions imposed by Appendix K. The conditions are (1) the case when the hot spot is cooled by FLECHT-based two-phase heat transfer coefficients and (2) the case when the hot spot is cooled by rod-to-rod radiation and conservatively derived steam cooling heat transfer coefficients. This variation in PCT (50°F) is estimated to be a factor of 10 to 20 times less than the PCT conservatism measured by best estimate analysis models.

C-E has performed best-estimate large break LOCA calculations for its standard 3800 MWt plant (References 1-3). These calculations used versions of the best estimate model at different times during it's development. Model benchmarking and verification was in part accomplished by comparisons with integral tests on such facilities as Semiscale and LOFT. References 2-4 demonstrate the excellent correspondence of C-E's pre-test predictions for LOFT Test L2-3. This same model when applied to a typical PWR operating at its licensed limit, i.e. PCT of 2200°F, produces a peak cladding temperature of 1000 to 1200°F. This is consistent with the NRC views on peak cladding temperature (PCT) conservatisms as stated in SECY-83-472 (Reference 5). Hence, the EM variation in PCT with axial power distribution is much less than the PCT margin between the EM and best estimate model prediction.

References

1. "A Best-Judgement Analysis of Emergency Core Cooling System Performance," Kessler, T. and Fader, G. (C-E) Nuclear Technology, V.34, July 1977.

2. "C-E Application of LOFT Program Results," G. Menzel (C-E) presented at ANS Summer Meeting, Atlanta, June 6, 1979.

3. ACRS 260th General Meeting, December 10, 1981, Subject: C-E Application for Final Design Approval of CESSAR-FSAR.

4. "LOFT L2-3 Test Analysis", A. E. Scherer (C-E), LD-79-029, May 8, 1979.

5. SECY-83-472, "Emergency Core Cooling System Analysis Methods", November 17, 1983.

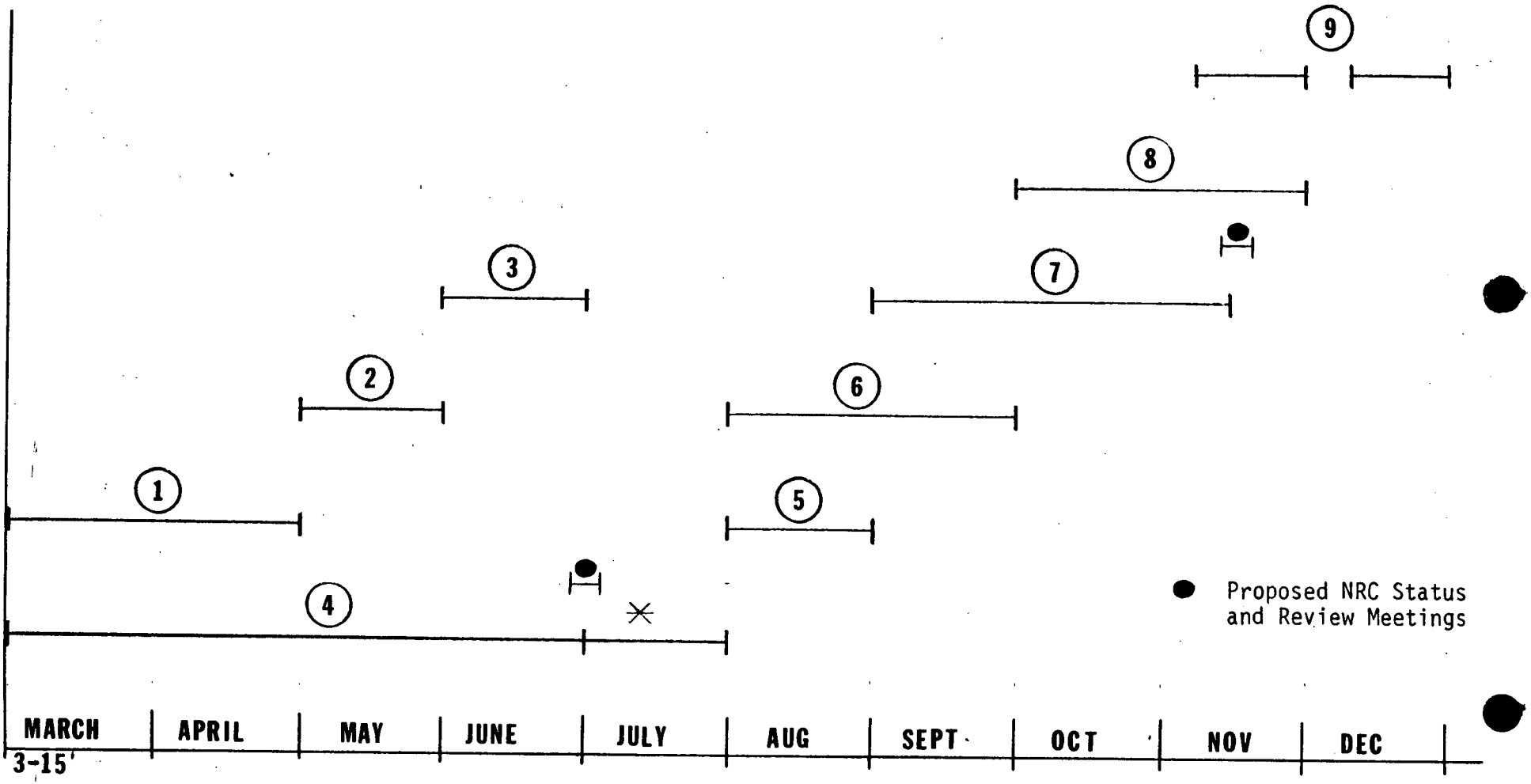
Attachment 3 to WPSC's Exemption Request

March 29, 1985

Reference 12

Schedule for Reanalysis to Confirm K(Z)

PROPOSED SCHEDULE FOR KEWAUNEE LOCA/ECCS REANALYSIS (WITH CURRENT MODEL CONSERVATISMS REMOVED) TO CONFIRM K(Z)



1. System parameters finalized (conservatisms removed) for revised LOCA/ECCS analysis.
2. Rebalance Primary System using Revised Plant Data.
3. Perform Limiting Break Blowdown Analysis.
4. Develop Revised Reflood Heat Transfer Correlations (RHTC), documentation transmitted to NRC for review and approval.
5. Incorporate Revised RHTC into REFLEX and TOODEE2, perform code verification.
6. LOCA/ECCS Exposure Sensitivity Analysis.
7. Revised K_Z Curve
8. Perform QA and issue Preliminary Report to WPS for review and comment.
9. Final Report incorporating WPS comments issued.

*The above schedule allows for a one month review of the RHTC by the NRC, if more time is required the schedule will be adjusted day-for-day.

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