As Revised 4/21/83

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket No. 50-454-OLA
	)	50-455-OLA
(Byron Station, Units 1 and 2)	)	

TESTIMONY OF THOMAS F. TIMMONS CONCERNING STEAM GENERATOR TUBE INTEGRITY (FLOW-INDUCED VIBRATION PHENOMENON)

Submitted on behalf of
the Applicant, Commonwealth Edison
Company, in Response to DAARE/SAFE
Contention 9c and League Contention 22

February 25, 1983

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NUCLEAR REGULATORY COMMISSION

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SUMMARY

The testimony of Mr. Thomas F. Timmons addresses the flow-induced vibration phenomenon and its relation to tube wear in Westinghouse Model D steam generators. After detailing his professional qualifications as an expert, Mr. Timmons describes the comprehensive program established by Westinghouse for the purpose of understanding and resolving tube wear problems resulting from flow induced vibration. The program, based on the collection of extensive operating plant data and laboratory and model test data, established that (i) tube vibration was less significant in Model D4 and D5 steam generators as compared to Model D2's and D3's, (ii) no significant tube wear is expected in Model D4 and D5 steam generators as long as main feed flow rates do not exceed 70%, (iii) modifications should be make to the Byron Station steam generators to permit the plant to operate at full power without any significant tube vibration.

Mr. Timmons testifies that Westinghouse has recommended to Commonwealth Edison Company that the preheater sections of the Byron steam generators be modified by expanding approximately 100 tubes at certain baffle plate locations and by diverting 10% of the feedwater flow through the auxiliary feedwater nozzle. Westinghouse expects to install the modifications during the third quarter of 1983. Mr. Timmons concludes that, as modified, no significant tube wear will be experienced in the Byron steam generators due to flow-induced vibration.

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TESTIMONY OF THOMAS F. TIMMONS CONCERNING STEAM GENERATOR TUBE INTEGRITY (FLOW-INDUCED VIBRATION PHENOMENON)

Q.1. State your name, address and present occupation.

A.1. My name is Thomas F. Timmons. My business address is P.O. Box 355, Pittsburgh, Pennsylvania 15230. I am employed by Westinghouse Electric Corporation as the Manager of Reactor Coolant Systems Components Licensing in the Nuclear Safety Department of the Nuclear Technology Division.

Q.2. State your educational background and professional work experience.

A.2. I graduated from Marquette University in 1968 with a Bachelors degree in Mechanical Engineering. Upon graduation, I received a commission as an Ensign in the U.S. Navy and was assigned to the U.S. Navy Nuclear Power Program. From August 1968 to August 1969, I successfully completed the courses of study at the U.S. Navy Nuclear Power School at Mare Island, California and at the U.S. Navy Nuclear Power Prototype School at Idaho Falls, Idaho.

From September 1969 to September 1972, I was assigned to the Engineering Department of the USS Bainbridge (DLGN-25), a nuclear-powered, guided-missile frigate. I served as the Reactor Laboratories Officer in charge of technicians who controlled reactor coolant system and steam generator chemistry and all radiological monitoring and control for the twin nuclear reactor plants of the USS Bainbridge. My subsequent duties included serving as the Electrical Officer in charge of the ship's electrical generation and distribution systems and components, including the electrical systems and components of the nuclear power plants. I was also qualified as Engineering Officer of the Watch (EOOW) and as Engineering Duty Officer (EDO). As EOOW, I supervised the operation and maintenance of the nuclear reactor plants while they were in operation and as EDO, I supervised the Engineering Department, including operation and maintenance of the nuclear reactor plants while the ship was in port. Following my separation from the U.S. Navy as a Lieutenant, senior grade, I took a three month vacation.

From January 1973 to August 1975, I was employed by the WEDCO Corporation, a subsidiary of Westinghouse Electric Corporation, at the Indian Point Nuclear Station as an Electrical Startup Engineer in the Operations Department. My duties included reconciliation of as-built conditions with drawings, post-installation electrical checkout of control circuits, motors, circuit breakers, etc. and supervision of startup testing.

From September 1975 to January 1980, I was employed in the Safety Standards Group in the Nuclear Safety Department of the PWR Systems Division of the Westinghouse Electric Corporation located at Monroeville, Pennsylvania. There I held positions as an engineer and as a senior engineer. My duties included development, evaluation and application of safety criteria in safety evaluations of nuclear power plant components and systems. During 1978 and 1979, I also coordinated a research program on nuclear power plant operator response to accident situations.

In February 1980, I was appointed Manager of Mechanical and Fluid Systems Evaluation in the Nuclear Safety Department. In this capacity, I was responsible for licensing activities and safety evaluations in the areas of fluid systems and mechanical components, including steam generators, for operating and non-operating plants with Westinghouse Nuclear Steam Supply Systems. In March of 1982, I was appointed Manager of Reactor Coolant Systems Components Licensing, a position that I currently hold. In this position, I am responsible for all licensing activities and safety evaluations for the reactor coolant system and its components, including steam generators, for operating and nonoperating plants. Since March 1982, I have also been assigned a collateral position as Manager, Licensing for the Model D Steam Generator Task Force.

Q.3. What are your responsibilities as Manager, Licensing for the Model D Task Force?

A.3. As Manager, Licensing for the Model D Steam Generator Task Force, I am responsible for the licensing support activities necessary for resolution of the flow induced vibration issue. These activities include the collection and review of engineering information; the performance and documentation of safety evaluations; the definition and evaluation of applicable regulatory criteria; coordination, review and submittal of licensing reports; and management of licensing interactions with utilities and their nuclear regulatory agencies.

Q.4. What is the purpose of your testimony?

A.4. My testimony addresses Rockford League of Women Voters' Contention 22 and DAARE/SAFE Contention 9(c) insofar as these contentions concern the potential for steam generator tube wear resulting from flow induced tube vibration in the preheater section of Westinghouse Model D steam generators.

Q.5. What design variations exist within Model D steam generators?

A.5. There are four models, or variations of Model D steam generators, called Models D2, D3, D4 and D5.

An outline drawing of a preheat steam generator is given in attachment 1. As shown in this figure, the preheat region is located on the cold leg side of the tube bundle and faces the feedwater inlet nozzle. Within the Model D preheat steam generator series, there are two general types of steam generators. These are the split-flow type (Models D2 and D3) and the counterflow type (Models D4, D5). These two types of steam generators differ significantly in the configuration of the inlet nozzle water box area and in the flow paths for the feedwater in the preheater itself.

In the split-flow type (see attachments 2 and 3), the incoming feedwater enters at the midsection of the preheater, encounters a circular impingement plate, which directs the flow outward over the front of the outer row of tubes, and then the feedwater enters the tube bundle. At the rear of the inlet pass (at the centerline of the steam generator) the flow "splits" with portions being directed upward and downward around the tubes and baffles. Model D3 has a different impingement plate design than the D2.

In the counterflow type (see attachment 4), the incoming feedwater enters the inlet water box and impinges on a wall that directs the water outward to fill the water box volume and downward to the preheater inlet pass located near the bottom of the steam generator. The water enters the tube bundle at the inlet pass, flows around the tubes and then upward around the tubes and baffles. This upward flow is "counter" to the direction of the flow of the primary water inside the steam generator tubes. Models D4 and D5 are also equipped with a "T"-shaped blowdown pipe to minimize the accumulation of sludge on the tubesheet. This blowdown pipe is accommodated by a "T"-shaped lane (see attachment 5). The Model D5 steam generator differs from the D4 in that it has ferritic stainless steel baffles and support plates.

Q.6. How do these major design features affect tube vibration?

A.6. In the split-flow type steam generator, the incoming feedwater encounters the impingement plate, is directed outward over the front of the outer row of tubes and then turns to enter the tube bundle. At full flow conditions, this directing and turning of the feedwater flow can produce vibration of some tubes in the front four rows of tubes. The differences in D2 and D3 impingement plate design affect only the local flow distribution patterns.

In the counterflow type steam generator, the incoming feedwater is distributed within the inlet waterbox before it enters the tube bundle. Because of the different distribution of the incoming feedwater over the front of the tube bundle, less vibration of the tubes in the outer row of the tube bundle and some tubes in the next two rows of tubes occurs. The presence of the "T"-slot in the tube bundle produces a pathway for flow to preferentially enter and permit some amplitude of vibration of a few additional tubes on either side thereof. The amplitudes of vibration of tubes in the counterflow models are less pronounced than those observed in the split-flow models. The main feed flow rate at which tube vibrations become significant is higher for the counter flow models than it is for the split-flow models.

Q.7. What is the origin of the flow induced vibration issue in Westinghouse preheat steam generators?

A.7. Preheater region tube wear was initially identified in Sweden at Ringhals Unit 3, a plant with Model D3 steam generators. On October 21, 1981, the plant was shut down due to an approximate 2.5 gpm primary to secondary leak. Subsequent examination revealed a through-wall hole in a single steam generator tube at a baffle plate. That tube was located in the outer row of the tube bundle facing the feedwater inlet. Eddy current testing (ECT) of the steam generator tubes indicated tube wall wear in some tubes located in the outer rows near the inlet nozzle. ECT was then performed at Almaraz 1, a nondomestic plant with D3 steam generators; at McGuire 1, a domestic plant with Model D2 Steam generators; and at Krsko, a non-domestic plant with D4 steam generators. The ECT of Almaraz 1 also revealed indications of possible tube wear in tubes located in the outer rows of the bundle. The plants with Model D2 and with Model D4 steam generators had no indications of possible tube wear. These two plants had not operated above 50% power. Based on these early ECT indications, Westinghouse established an extensive program to determine the cause and extent of tube wear in Model D steam generators.

Q.8. What is included in this program?

A.8. Westinghouse has undertaken an extensive program to investigate, understand and define vibration and tube wear in Model D steam generators and to conceive, develop, test and evaluate any modifications necessary to allow operation of Model D steam generators at full power. This program includes gathering, reviewing and analyzing data from operating plants and from laboratory and model tests.

Q.9. Can you tell us more about the operating plant data collection and analysis portion of the program?

A.9. Yes. Operating plant data was obtained from four of the five operational plants with preheat model steam generators. A fifth plant, Angra. 1, was not used as it was undergoing startup testing and low power operation and had not operated with any flow through the main feedwater nozzle. Likewise, two additional plants with preheat model steam generators that began their initial startup program at the end of 1982 were not included.

The collection of operating plant data began in October 1981 with the eddy current testing (ECT) performed at the Ringhals 3 plant and at the other operating Model D plants. For the two plants with D3 steam generators that had operated at 75% power or above, the ECT showed indications at baffle plate locations in the outer rows of the preheater near the feedwater nozzle. Visual and metallurgical examination and analysis of two steam generator tubes removed from the outer row of the preheater region of the Ringhals 3 plant revealed the presence of wear marks at the baffle plate locations. For those Model D plants that had operated at power levels of 50% or less, no ECT indications of tube wear were observed. Based on these data, in November 1981, Westinghouse recommended 1500 hours as an interim operating period and 50% power as an interim, maximum operating power level.

In December 1981, two tubes were removed from and tube instrumentation was installed in the preheater region of the Model D3 steam generators at the Almaraz 1 plant. Upon resumption of operations in January 1981, vibration data was obtained at various power levels. The Krsko (Model D4) and McGuire (Model D2) plants operated at 50% power for approximately 1500 hours before another ECT inspection was performed and vibration instrumentation was installed. The ECT data from Krsko and McGuire did not reveal indications of tube wear. Upon startup of these plants, vibration data was obtained at various power levels and feedwater configurations. This additional ECT data and the tube vibration data were reviewed and analyzed by Westinghouse.

The aforementioned data from operating plants were used to determine the extent of tube vibration and to determine the main feed flow rates below which significant tube vibration would not be expected to occur. It was observed that operation at or below 50% for Model D steam generators did not produce any significant tube vibration or any significant changes in tube wear. It was also observed that operation of the Krsko steam generators at or below 70% main feed flow did not produce any significant tube vibration. Based on the plant data and laboratory test data available at that time, Westinghouse continued to recommend 1500 hours as a prudent interim operating period, and continued to recommend a maximum of 50% main feed flow as an interim feed flow for Models D2 and D3 and a maximum of 70% main feed flow for Model D4.

In March of 1982, the Almaraz 1 plant shut down after approximately 1500 hours of operation at 50% power and performed an ECT inspection of the steam generators. No significant changes in the ECT signals from the previous inspection of tubes in the preheater were observed. At that time, two tubes were removed from the steam generators at Almaraz 1. Visual and metallurgical examinations and analyses of these tubes revealed the presence of wear marks at baffle plate locations. These wear marks were similar to those observed on the tubes removed from Ringhals 3 in November 1981. Additional ECT inspection was performed at Ringhals 3 and an additional tube was removed for visual and non-destructive examinations. From the ECT data, it was determined that the results of the eddy current testing performed in October of 1981 over-estimated the depth of wear at baffle plate locations. A new method of eddy current testing was qualified by performing ECT testing on the removed tube prior to its removal and after its removal from the steam generator. This new method of eddy current testing provides more accurate data on the depths of tube wear marks. In April of 1982, the Ringhals 3 plant resumed operation at 40% power. Data from vibration instrumentation installed in this plant was similar to that taken from other split flow steam generators.

During the period from November 1981 until May 1982, Westinghouse performed further analyses, evaluation and correlation of available data from removed tubes, ECT, vibration instrumentation and laboratory testing. As a result, Westinghouse developed a certain empirical data base with respect to operation of Model D steam generators. This data base permitted Westinghouse to make conservative estimates of potential tube wear for operating conditions extending to 100% main feed flow. During the last half of 1982, the McGuire 1 plant was operated for two periods of approximately 700 hours each for a total of approximately 1400 hours at a power level of 75%. At the end of each period, the ECT indications observed in the McGuire steam generators were, with the exception of one tube, within the bounds of the wear projected by Westinghouse.

In May 1982, the Krsko plant performed an ECT inspection, installed additional vibration instrumentation, and removed one tube from a steam generator for visual and metallurgical examination and analysis. No indications of tube wear were detected by ECT. The removed tube had some wear with a depth below the limit of ECT detectability. Modifications to the feedwater bypass system to allow operation of the plant at 100% power with up to 30% of the feedwater flow bypassing the preheater of the steam generator were also installed at this time.

After resumption of operations, vibration data was collected at various power levels and various combinations of main feed and bypass feed flows (e.g. 70/0, 70/30, 90/10, 100/0, etc.). From this data it was observed that the tube vibrations at the 70% main feed/30% bypass feed combination were slightly greater than those at the 70/0 combination and that the vibrations observed at 70/30 were acceptable.

In November 1982, the Krsko plant performed an ECT inspection, installed additional tube vibration instrumentation in both steam generators, removed two tubes and expanded one tube at baffle plate intersections. No indications of tube wear were observed from the ECT inspection. The two removed tubes had wear marks of .001 to .002 inches in depth which are below the limit of ECT detectability. After resumption of operation, tube vibration data were obtained. From this data, it was observed that the tube vibrations in both steam generators were similar and had not changed with time. The expanded tube had been previously instrumented for vibration and was re-instrumented. Previous tube vibration data was compared with the data obtained after the tube had been expanded and it was concluded that the tube vibrations were reduced by at least a factor of 5 from the non-expanded case. Based on conclusions from the data base and on ECT data, Westinghouse recommended that the operating interval at 70/30 be increased to approximately 4500 hours.

In addition to the previously mentioned uses of the tube vibration and removed tube data, these data have also provided baseline information that was used to calibrate and qualify laboratory test models and the flow-induced-vibration dynamic analysis model.

Q.10. Please discuss the laboratory and model tests that were conducted as a part of the Westinghouse program.

A.10. Various size scale models of the steam generator preheater region have been constructed and have provided data not obtainable from operating plants. Since two significantly different preheater flow designs comprise the Model D series steam generators, two subprograms were undertaken for the laboratory tests: one for the Models D2 and D3 (split-flow) steam generators and another for the Models D4 and D5 (counterflow) steam generators.

For the split-flow steam generators, a 0.417 scale model, a 2/3 scale model and the Swedish (SSPB) full scale model were utilized. The 0.417 scale model consisted of the full preheater region and contained the full complement of tubes. All flow passes within the preheater were modeled. Tube vibration data was the main output from this model, by using tubes instrumented internally with strain gages. Additionally, some flow velocity data was obtained in areas of interest.

The 2/3 scale model included all components (i.e., impingement plate, tubes, baffles and supports) from the feedwater nozzle to the exit of the first pass. All tubes were present with instrumentation provided within the first five rows of tubes nearest to the feedwater nozzle. This test model was designed to measure shell-side local water velocities and to determine the flow distribution patterns within these first rows of tubes. Additionally, the model permitted the measurement of steady state and oscillating flow-induced drag forces on these tubes.

The SSPB full scale model duplicates the inlet pass of the preheater and includes a section of full length tubes representing the full height of the tubes from the tubesheet to the U-bend elevation. Adjustment of the alignment of tube support plates provided for simulation of the actual hot operating steam generator support plate conditions as well as the replication of tube vibration to compare with actual operating plant measured tube vibration.

A similar program encompassing various test models was established for the counterflow steam generator program. Here, a 0.95 scale air model, a 1/4 scale water model, and a 16° full scale water model were used. Additionally, the 2/3 scale water model from the split-flow program was modified for testing in the counterflow configuration.

The 0.95 scale air model was used to determine flow velocity distributions within the preheater. The flow distribution patterns obtained from this model were then verified in the 2/3 scale water model. In addition to determining the shell side local water velocities and flow distribution patterns, the 2/3 scale water model permitted measurement of the drag forces on the tubes and the pressure drops at various locations within the preheater.

A 16° full scale model was used to replicate in the laboratory the tube vibration response observed in operating steam generators. This model consists of one half of a preheater region (the other half is symmetrical). Within this region, all of the tubes were installed with those tubes contained within a 16° "slice" being full length (up to the U-bend elevation). This model, like the SSPB model, was used to test various tube/support plate interactions under varying inlet flow velocities and distributions.

A single tube vibration model was used to characterize tube response under various excitation and support conditions. Here, a device was used to vibrate a full length tube. Support plates were located at the same elevations as in the actual steam generator.

The use of these various test models provided the additional capability of testing various concepts designed to reduce tube vibrations. By this manner, several design concepts were rejected while others are being optimized. Testing to optimize concepts and to obtain data on performance will be completed in the second quarter of 1983.

Concurrent with collecting and analyzing operating plant and laboratory test data, a computer model was developed to predict tube behavior as a result of flow induced vibration. This model is a multi-span dynamic analysis model which uses annular gap elements at the support plate locations and is thus able to simulate tube response within the support plate clearance. The gap elements can be offset to simulate various support conditions. Results from this model have been correlated with data obtained from operating plants and scale models.

Consideration was also given to installation of proposed modifications within the actual steam generators. Full scale mockups of the steam generator preheater region to include the feedwater nozzle and surrounding obstacles were constructed. The ability and ease of installation of each proposed modification was evaluated. The mockups also provided a means to test installation tooling and to train maintenance personnel.

Q.11. What conclusions were reached by Westinghouse as a result of the Model D program described in answer to Q.9. and Q.10.?

A.11. Evaluation of the data from the Model D program reduced to date has provided an insight into the unexpected phenomenon which has produced the tube wear occurring within the preheater region of Model D steam generators. Conclusions may be drawn that for the split-flow steam generators in the unmodified condition (Models D2 and D3), operation at high main feed flow rates may produce significant tube vibration in the outer rows of tubes closest to the inlet of the preheater. This vibration can produce tube wear.

Operation of Models D2 and D3 steam generators in the unmodified condition at main feed flow rates up to 50 percent has not produced any significant tube vibration or tube wear. Installation of the D2/D3 modification should permit the plant to operate at 100 percent main feed flow without an increase in the potential for excessive tube vibration or excessive tube wear.

Operation of Model D4 steam generators at high main feed flow rates could produce significant tube vibration of a few tubes. Since the anticipated vibration is less pronounced than that observed in operating Model D2 and D3 steam generators, any wear that may occur in Model D4 steam generators attributable to tube vibration is expected to be less.

To date, operation of Model D4 steam generators with main feed flow rates up to 70 percent has not produced any tube wear that can be detected by ECT, although visual examinatin of three removed tubes did disclose some small amount of wear, approximately 0.001 to 0.0025 inches in depth.

The feedwater bypass modification installed at Krsko has been effective in reducing tube vibrations to low levels and has permitted the plant to operate at full power.

Q.12. Is the Krsko tube vibration experience applicable to the steam generators installed at the Byron Station?

A.12. Yes. However, since the feed flow at Byron is 7% lower than that at Krsko, the level of vibration is expected to be less.

Q.13. Will Westinghouse recommend a modification of the Byron plant to minimize tube wear from flow-induced vibration?

A.13. Yes, Westinghouse has recommended that Commonwealth

Edison Company make modifications to the Byron plant to reduce the potential for significant tube vibrations in the Byron steam generators. These modifications are: 1) the expansion at baffle locations of approximately 100 tubes per steam generator and 2) the bypassing of approximately 10 percent of the flow from the main feedwater nozzle to the auxiliary feedwater nozzle. The expansion of tubes at baffle plate locations will limit the tube movement at the baffle plate intersections to a few thousandths of an inch. The bypassing of 10 percent of the main feed flow to the auxiliary nozzle of the steam generator will reduce the main feed flow at the inlet to the preheater to approximately 90 percent and will further reduce the potential for vibration of the tubes in the preheater.

Q.14. How will expansion of the tubes be accomplished?

A.14. Westinghouse has developed a proprietary process that will be used to expand the steam generator tubes. The process involves the insertion of tools into the tubes from the primary side of the steam generator tubesheet. The tools are then used to locate the baffle plate intersection and to expand the tube at the appropriate location. The expansion zone will be entirely within the thickness of the baffle plate. (Baffle plates are provided within the preheater section of the steam generator to direct the flow past the tubes.) After the expansion has been effected, the expansion is verified by the use of Eddy Current Testing.

Q.15. How has Westinghouse evaluated the effect of the tube expansion on the integrity of the steam generator tubes?

A.15. Expansion of tubes in steam generators has long been utilized in the manufacture of steam generators. Westinghouse established a program to evaluate the effect of tube expansion. This program included an evaluation of the levels of residual stresses in expanded tubes. Westinghouse has concluded that the levels of residual stresses in the expanded tubes, coupled with the relatively low temperature in the preheater region, does not significantly increase the potential for stress corrosion cracking in the expanded location.

Westinghouse has also conducted accelerated corrosion testing to assess the effects of the reduced tube-to-tube hole clearance on the potential for denting of the expanded tubes. The results of this testing indicates that the potential for denting is not increased for tubes expanded at the baffle intersections.

Westinghouse has also performed structural analyses of the expanded tube for design basis, transients and accidents. The results of the struct ural analyses indicate that the ASME Code allowable values for stresses and fatigue usage factors are not exceeded for expanded tubes.

Q.16. How will the feedwater bypass modification be accomplished for Byron?

A.16. For the Byron plant, the feedwater bypass modification will require that the present feedwater preheater bypass valve remain open during high main feed flow rates. This will result in approximately 90 percent of the feedwater flow entering the main feedwater nozzle and the remainder of the feedwater flow entering the steam generator through the auxiliary feedwater nozzle.

Q.17. What effect will the proposed modifications have on tube vibration?

A.17. Westinghouse has tested the modifications in the 16° Model and in the KRSKO plant. In the 16° Model, a number of tubes were expanded and testing was conducted to determine the effect of tube expansion on tube vibration. At a flow rate equivalent to 90 percent of the Byron main feed flow rate the expanded tubes exhibited vibration levels that were less than those observed at flow rates equivalent to 70 percent of the Byron main feed flow rate without tube expansion. As I indicated earlier, a 70% main feed flow rate will not result in significant tube wear. In addition to the testing in the 16° Model at the krsko plant, one tube that had been previously instrumented was expanded at baffle plate locations. Previous tube vibration data was compared with the data obtained after the tube had been expanded and it was concluded that tube vibrations were reduced by at least a factor of 5 from the non-expanded case. This reduction resulted in a negligible level of vibration for that tube.

In my opinion, these model and test data demonstrate that flow-induced vibration in the Byron steam generators will be minimized to the point where tube wear will not significantly affect the structural integrity of the Byron steam generator tubes.

Q.18. When will a modification be available for installation at the Byron Station?

A.18. The present schedule for the engineering program is expected to have the modification available for installation in the Byron Station by the third quarter of 1983 to support plant schedules.